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Scoping Assessment of the Environmental Health Risk Associated with Accidents in the LWR Supporting Fuel Cycle

Teknekron, Inc, Washington, D C

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SCOPING ASSESSMENT OF THE ENVIRONMENTAL HEALTH RISK ASSOCIATED WITH ACCIDENTS IN THE LWR SUPPORTING FUEL CYCLE

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ABSTRACT

A framework is developed for the assessment of the health risk associated with postulated accidents in the fuel cycle supporting the annual operation of a 1000 Mwe light water-cooled reactor. The methodology applied consists of a synthesis of nominal radiological source terms and corresponding likelihoods for postulated accidents at generic fuel cycle facilities (with the exception of the reactor and waste management) considered to comprise the nuclear industry toward the end of the current decade. The accident definitions, source terms, and likelihoods are compiled from a number of diverse studies and subjected to interpretation, renormalization, and, in some cases, revision. Risk is defined in this study as the product of the expected consequences of an accident and its likelihood of occurrence. The source terms are converted to 50-year dose commitments following, in general, the recommendations of the International Commission on Radiological Protection, and considering inhalation,dietary, plume submersion and ground shine pathways to man for generic representations of population density. Dose commitments are converted to health effects using the linear, non-threshold dose-response relationship developed in the BEIR report (Report of the Advisory Committee on the Biological Effects of Ionizing Radiation) and interpreted by EPA.

An initial quantitative estimate of the aggregated somatic health risk from accidents in the supporting fuel cycle, normalized to the annual operation of a 1000 Mwe LWR, is of the order of roughly 10^{-3} excess cancers. This estimate is subject to uncertainties associated with the dose conversion model, the dose-response relationship, and the consequence expectation value estimate. The results, however, are broken

down by accident category in each component of the fuel cycle, so that the framework thus established can be expanded or revised as more data or the results of more refined analyses become available.

Transportation accidents dominate the total accident risk, whereas the risks from accidents in mining, milling, and plutonium storage are relatively insignificant. The risks from the remaining components of the fuel cycle, with the exception of spent fuel reprocessing, are roughly of equal orders of magnitude, at least at the upper limits of the range of estimates. Comparisons with the risks from normal operations of the supporting fuel cycle and with occupational risks indicate that, on the basis of the annual operation of a 1000 Mwe LWR, the risk from accidents in the supporting fuel cycle is orders of magnitude lower. On the same basis, the risk from accidents in the supporting fuel cycle is also slightly lower than that from reactor accidents and comparable to that from normal reactor operation.

A more comprehensive scoping analysis would include the risks associated with site-induced and other high consequence, low probability ("class 9") accidents, which were, in general, neglected here. For example, the likelihoods and consequences of accidents at the reprocessing facility involving the interim storage of high level wastes should be evaluated. Nor have the risks associated with proposed and postulated waste management alternatives been addressed, although this area is the subject of an extensive effort currently sponsored by the Energy Research and Development Administration. Moreover, the accidental releases of chemicals at fuel cycle facilities have the potential of producing environmental health effects. Finally, more confidence could be placed in these results if a more detailed assessment of HEPA filter failure probabilities were available, and if accident scenarios associated with spent fuel reprocessing and transportation were examined in considerably more detail.

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1. INTRODUCTION

1.1 Background

Since the advent of the National Environmental Policy Act of 1970 (NEPA),[1] the perspective in the assessment of environmental health effects associated with nuclear power production has shifted significantly. Historically, the radiological releases from normal operations of nuclear reactors were evaluated against maximum individual dose limitations at the site boundary. Accidental releases were evaluated using conservative assumptions regarding release magnitudes and environmental dispersion.* No serious attempts were made to estimate the likelihood of accidental releases. Moreover, the evaluation of the environmental health effects associated with the nuclear fuel cycle in support of reactor operation received little attention, either during normal operations or in the accident mode.

The mandate of NEPA resulted in a number of studies which assess the environmental health effects associated with nuclear power generation on a realistic basis. These assessments are now performed routinely in the form of Detailed Environmental Impact Statements as part of the licensing process associated with the construction and operation of each reactor or individual fuel cycle facility. Moreover, a number of generic studies have been performed in support of impending legislation or regulatory rule-makings. For example, the environmental statement in support of Appendix I to 10 CFR 50,[4] which established numerical guides to meet the "as low as practicable" criterion,evaluated population doses and potential health effects resulting from the normal operation of reactors assuming various levels of control technology. For the supporting nuclear fuel cycle, the GESMO (Generic Environmental Statement on Mixed Oxide Fuel) Report[5] evaluated the magnitude of effluents, population doses, and potential health effects for each component of the fuel cycle both with and without plutonium recycle. The consequences and likelihoods of major reactor accidents were predicted on a realistic basis in the recent Rasmussen Report.[6]

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* Examples of studies utilizing this approach are WASH-740[2] for nuclear reactors and ORNL-3441[3] for fuel fabrication and processing plants.

Potential accidents in the supporting nuclear fuel cycle have received only cursory evaluation. A survey of fuel cycle accidents and their potential consequences was incorporated in the Environmental Survey of the Uranium Fuel Cycle,[7] but this study did not attempt an evaluation of accident likelihoods or health risks. Much of the accident information incorporated in this study was based upon data contained within detailed environmental reports and statements for individual fuel cycle facilities, each of which contains an accident assessment based upon realistic assumptions.

Accident likelihoods have been assessed on a generic basis for the mixed oxide fuel fabrication,[8] spent fuel reprocessing,[9] and transportation[10] components of the fuel cycle. Different methodologies were used in these assessments, and the results are not readily translatable to health risks. None of the previous accident evaluations has normalized the predicted consequences to the operation of the reactor supported by the fuel cycle.

1.2 Objectives and Scope

The objective of the current study is to provide a framework for the risk evaluation of postulated accidents in the nuclear fuel cycle, and to establish a preliminary quantitative estimate of this risk. The estimated risk will be normalized to the annual operation of a generic reactor so that the impact of potential fuel cycle accidents can be incorporated in the overall cost/benefit balance for nuclear power generation.* The results will be broken down by accident category in each component of the fuel cycle, so that the framework thus established can be expanded or revised as more data or the results of more refined analyses become available. The expanded format will also be useful in identifying relatively high risk operations in the fuel cycle. Finally, the overall risk from accidental releases will be compared with the risk from normal operation of the reactor and the supporting fuel cycle.

The analysis will be confined to the light-water-cooled reactor (LWR) fuel cycle incorporating the recycle of both uranium and plutonium. All components of the conventional LWR fuel cycle will be included with the exception of waste management, which is omitted because of uncertainties in process definition. Although accidents in the fuel cycle may involve the release of toxic chemicals to the environment, only the effects of radiation on human health (somatic effects) will be considered. Similarly, occupational risks and the risks to the public from deliberate acts of sabotage or diversion of nuclear materials are considered outside of the scope of the current study.

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* The most efficacious way to express the social benefits of nuclear fuel cycle activities in support of reactor operation is in terms of the electricity delivered by the reactor.

1.3 Methodology

Risk is defined in this study as the product of the expected consequences of an accident and its likelihood of occurrence. The accident risk in a component of the fuel cycle is obtained by summing the risks from all postulated accidents for a generic facility representing that component. The risk from each component is normalized to the annual operation of a 1000 MWe generating station using mass flow data derived for a generic LWR. The risk from accidents in the entire fuel cycle is then obtained by summing the normalized risks from the individual components of the fuel cycle.

The heart of this study lies in the definition of postulated accidents, source terms (compositions and magnitudes of radiological releases to the environment) associated with these accidents, and likelihoods of their occurrence. In most cases, this information has been synthesized from existing studies of considerably less ambitious scope. Individual items of data, however, have been subjected to considerable manipulation, interpretation, and renormalization. Accident likelihoods, in particular, are based upon a diversity of original methodologies, including actual incidents on record (lacking, unfortunately, in comprehensiveness), analogous chemical industry statistics, and fault tree analysis. Ranges are quoted in a number of instances, and judgment enters in nearly all instances. The lack of rigor is hopefully compensated by a thorough documentation of original sources.

The source terms are converted to doses generally following the recommendations of the International Commission on Radiological Protection

(ICRP). Inhalation, dietary, plume submersion, and ground shine pathways are considered in deriving 50 year dose commitments. Then, assuming uniform population distributions surrounding each generic fuel cycle facility, a deposition model is used to determine the total population dose, in man-rem, to critical organs of humans in the environment. Similarly, a generic river model is defined to estimate the dilution of liquid sources in the watercourse. Population dose from liquid sources is then estimated from the drinking water pathway for the generic river.[11]

The population dose expresses the integral of the radiation doses received by individuals over the entire population exposed.[12] Coupled with the linear non-threshold hypothesis, this approach permits the direct conversion of population dose to health effects. According to this hypothesis, there exists a linear relationship between the total accumulated dose and the number of health effects from zero exposure to the highest exposure which does not cause acute mortality. This linear relationship is defined from estimates of somatic effects[12] originally obtained from the BEIR Report.[13]

1.4 Limitations

The rather simple concept of risk employed in this study is not intended to minimize the philosophical difficulties in defining risk or the complexities of applying the concept to decision-making.[14] The linear nonthreshold hypothesis for converting population dose to health effects is not universally accepted,[15] but has been generally adopted as a conservative approach for the establishment of radiation standards.[12] Moreover, the application of this linear model permits the normalization of risk to the annual operation of a generic LWR and generally simplifies the analysis.

As discussed in the previous section, the weakest link in the accident analysis lies in the establishment of source terms and likelihoods for the accidents postulated in each component of the fuel cycle. First of all, the comprehensiveness of the list of accidents postulated cannot be assured. Indeed, it will become apparent in Section 4 that this scoping assessment is not altogether complete. For example, the consequences of natural disasters and other "class 9" accidents have not, in general, been examined here. An analysis of the nature attempted here is a continuing exercise, thus explaining the early identification of this study within the context of a "framework."

The use of incidents on record to establish an accident data base is deficient in at least two respects. The existing data base is not comprehensive and does not, in general, contain estimates of environmental releases. Moreover, processes and safeguards have been improved in a number of specific cases, so that the historical record is not necessarily indicative of current or future events.

The use of analogous chemical industry statistics may be misleading because of differences in process constraints and the generally higher concern for safeguards in handling nuclear materials. The fault tree approach depends upon the ability to conceptualize all potential release events and suffers from the unavailability of a complete set of quantitative failure data.

In reality, a complete spectrum of source terms and corresponding probabilities is associated with each potential accident class. The choice in this assessment of "nominal"source term and a single likelihood associated with this release is a simplification. However, the nominal source terms and associated likelihoods have been chosen to be as representative of the actual continuum of sources as possible. The selections of source terms are based upon past analyses or actual data representative of process variables. In some cases, though, past experience is not completely representative of the processes and plant capacities selected for this study. Also, in some cases, it is not possible to tightly couple the accident likelihoods with the source terms.

Despite the foregoing limitations, it is felt that the synthesis presented here is a necessary first step in consolidating and placing in perspective previous estimates of the impact of accidents in the fuel cycle. The nature of the study and the many assumptions employed render the quantitiative estimates of risk highly tentative. A framework is developed, however, which can be used to update these estimates as more complete and accurate data become available.

1.5 Organization

Section 2 of this report provides a brief description of the LWR supporting fuel cycle conceptualized for this study, including the factors used in the normalization of the results to the annual operation of a generic 1000 MWe LWR. Section 3 describes the models used for the dispersion of radioactive sources, population distributions, dose conversions, and health effects.

The heart of the study is contained in Section 4, which provides estimates of the source terms and corresponding likelihoods, including documented rationale, for the accidents postulated in each component of the fuel cycle. The predicted population doses and normalized health risks corresponding to each postulated accident are compiled in Section 5. The results are consolidated to provide an estimate of the normalized health risk associated with the entire supporting fuel cycle, and compared with the risk associated with normal operations of the supporting fuel cycle and the reactor. Conclusions and recommendations for future work are contained in Section 6.

2. THE LIGHT-WATER-COOLED REACTOR (LWR) FUEL CYCLE

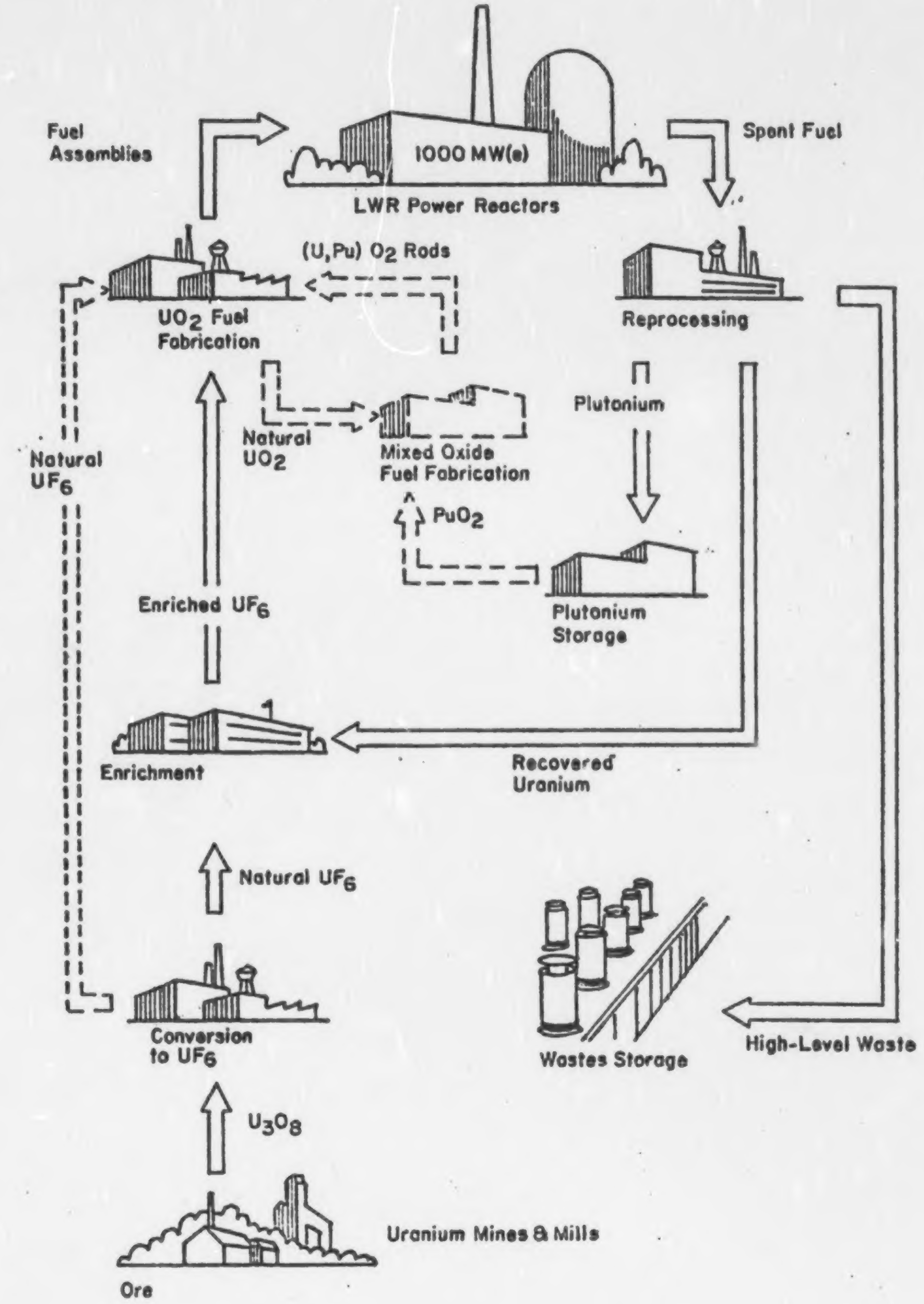
2.1 Description of the Fuel Cycle

The conventional LWR fuel cycle is shown schematically in Figure 2-1. The reactor and high-level waste management are included in the figure, although these components are not treated in the current study. The capacities of the generic facilities comprising the remaining elements, denoted as the LWR supporting fuel cycle, are given in Table 2-1.

The fuel cycle shown in Figure 2-1 assumes the complete recycle of the uranium and plutonium contained in the spent fuel. In the absence of plutonium recycle, the dotted elements in the figure are eliminated. Should spent fuel reprocessing also be eliminated from the fuel cycle, the spent fuel itself would be stored or disposed of, and plutonium storage, reprocessing, and the recycle of uranium to the enrichment plant would be eliminated in addition to the dotted elements.

The specific components comprising the LWR supporting fuel cycle are described in References 5 and 7. Summary descriptions are given below:

- Mining - The generic uranium mine is considered to be a surface mine in the Western United States. The nominal capacity is 480,000 MT ore/yr, a larger than average surface mine, situated on 3000 acres of land. The U_3O_8 content is assumed to be 0.2%, and the ore body lies at various levels from 100 to 450 feet below the surface. The ratio of overburden to ore body is estimated to average about 30 to 1. About 33 surface mines and 122 open pit mines (supplying 36% of the uranium)



Fuel Assemblies
1000 MW(e)
LWR Power Reactors
Spent Fuel
(U,Pu) O2 Rods
UO2 Fuel Fabrication
Reprocessing
Natural UO2
Mixed Oxide Fuel Fabrication
Plutonium
PuO2
Natural UF6
Enriched UF6
Plutonium Storage
Enrichment
Recovered Uranium
Natural UF6
Conversion to UF6
Wastes Storage
High-Level Waste
U3O8
Uranium Mines & Mills
Ore

Figure 2-1. The LWR Fuel Cycle

TABLE 2-1

CAPABILITIES OF THE GENERIC FACILITIES COMPRISING THE LWR SUPPORTING FUEL CYCLE

FACILITY	ANNUAL CAPACITY
Mine	480,000 MT ore
Mill	960 MT U_3O_8
UF_6 Conversion Plant	5,000 MTU
Enrichment Plant	8.75×10^6 kg SWU
Uranium Fuel Fabrication	900 MTU
Reprocessing	1,500 MTHM
Mixed Oxide Fuel Fabrication	300 MTHM
Plutonium Storage	40 MT Pu_f

MT = metric ton = 2205 lbs.

MTU = metric tons uranium

kg SWU = kilograms of separative work units

MTHM = metric tons heavy metal

$MTPu_f$ = metric tons fissile plutonium serviced the industry in 1973, and the output of uranium mines in 1990 is estimated to be roughly a factor of seven higher than current levels.[16]

- Milling - The milling step extracts the uranium in the ore and produces a refined U_3O_8 product. The generic uranium mill is located adjacent to the mine and produces approximately 960 MT U_3O_8 annually, which is roughly the average nominal capacity of mills currently in operation. It uses a mechanical crushing and acid leach, solvent-extraction process, which is the predominant chemical processing method. Approximately 17 uranium mills are currently in operation, and industry forecasts anticipate roughly 80 mills by 1990.[5]
- UF_6 Conversion - The U_3O_8 extracted from the uranium ore must be converted to a volatile compound, uranium hexafluoride, for enrichment by the gaseous diffusion process. There are currently two facilities in the U.S., each producing UF_6 by a different process. The dry hydrofluor process consists of continuous successive reduction, hydrofluorination and fluorination followed by fractional distillation for purification of the product. The wet process uses solvent extraction at the head end to prepare a high purity feed followed by reduction, hydrofluorination, and fluorination steps. The bulk of the impurities in the ore are rejected from the hydrofluor process as solids, whereas in the wet process, the impurities are contained in the raffinate stream. The generic UF_6 conversion plant produces 5000 MTU as UF_6 annually, and consists of a synthesis of the two existing

processes. By the year 1990, it is anticipated that approximately 5-6 plants of 15,000 MTU/yr capacity will be in operation.[5]

- Enrichment - The uranium used by the LWR must be enriched from the 0.7% ^{235}U content in the ore (and the ~0.8% ^{235}U content in the uranium recycled from the reprocessing step) to approximately 2-4% ^{235}U. The generic enrichment plant uses the gaseous diffusion technology which is the process adopted by the three existing government-owned and-operated enrichment plants. In the gaseous diffusion process, volatile UF_6 feed is compressed and pumped through hundreds of stages of porous barriers through which the gas molecules diffuse. In addition to the enriched UF_6 product, the gaseous diffusion process produces UF_6 depleted in ^{235}U, called tails. The generic enrichment plant has a capacity of approximately 8.75×10^6 Kg separative work units (SWU's) annually, larger than any of the three individual enrichment plants currently in existence, but typical of the new plants planned for the future. It is estimated that by the year 1990, approximately eight plants of this capacity will be required by the industry.[5]
- Uranium Fuel Fabrication - The enriched UF_6 is converted into fuel assemblies at the fabrication plant for use in the LWR. The predominant current method for uranium fuel fabrication is the ammonium diuranate (ADU) process for conversion of UF_6 to UO_2 powder, followed by pelletizing and sintering in a reducing atmosphere to achieve the required density. The finished pellets are loaded into

Zircaloy or stainless steel tubes, and the completed fuel rods are assembled in fixed arrays. Scrap uranium is recycled into the main process through a scrap recovery cycle. The preparation of the UO_2 powder or pellets may be carried out at a separate location from the final steps of the fabrication process. The generic fabrication plant is assumed to produce annually fuel assemblies containing 900 MTU , a large plant by today's standards. It is anticipated that roughly nine plants of somewhat larger capacity will be in existence by the year 1990.[5]

- Spent Fuel Reprocessing - Following a cycle of exposure in the LWR, the fuel is chemically processed to recover the unburned uranium and plutonium and separate the fission products for storage or ultimate disposal. The generic spent fuel reprocessing plant is located on a relatively large site and utilizes the Purex process to separate fissile material from the fission products. In this process, the fuel elements are chopped into short pieces, and the metal oxides are leached by hot nitric acid, leaving behind the chopped tubing. The nitric acid solution, containing uranium, transuranics and fission products, is adjusted chemically and processed through solvent extraction and ion exchange systems. These process steps separate the fission products, uranium, and plutonium from each other. The purified uranium product is converted to uranium hexafluoride and is shipped to the gaseous diffusion plant for reenrichment. The purified plutonium product is converted to PuO_2 for recycle to the mixed oxide fabrication plant. The high level liquid wastes are stored temporarily on-site in a water-cooled storage basin and ultimately converted to solid form for shipment to the waste-respository. The generic fuel reprocessing plant is assumed to have a capacity of 1500 MTHM/yr. Although there are currently no commercial reprocessing plants in operation in the U.S., approximately seven plants of the generic plant capacity are predicted to be in operation by the year 1990.[5]

- Mixed Oxide Fuel Fabrication - The extracted plutonium from the reprocessing step is combined with natural uranium at the mixed oxide fuel fabrication plant to form mixed uranium dioxide-plutonium dioxide fuel pins for recycle to the LWR.. In the conventional process, plutonium dioxide powder is blended with ceramic grade uranium dioxide powder, followed by the pelletizing, sintering, and mechanical steps employed in the uranium fuel fabrication steps. Scrap plutonium is recycled into the main process through a scrap recovery cycle. The fuel assemblies themselves may be fabricated from the loaded fuel rods at the uranium fuel fabrication plant. The generic mixed oxide plant is assumed to produce annually fuel assemblies containing 300 MTHM. Although the existing mixed oxide facilities are only pilot plant scale, it is anticipated that by the year 1990, approximately eight facilities of the generic plant capacity will be in existence.[5]

- Plutonium Storage - Plutonium recovered from the reprocessing step must be stored on an interim basis prior to use in mixed oxide fuel fabrication because of the difference between rates of production and use. Should plutonium not be recycled in LWR's, all of the plutonium recovered from LWR spent fuel would have to be stored awaiting the introduction of the breeder reactor or some alternative destiny. It is anticipated that the plutonium will be stored in the oxide form under quiescent conditions. The generic storage facility is assumed to hold approximately 40 MT of fissile plutonium.

- Transportation - Transportation implies the shipment of materials between each component of the fuel cycle described above, including the shipment of unirradiated fuel from fabrication to the reactor and irradiated fuel from the reactor to the reprocessing plant. It is assumed that all of these shipments, with the exception of the transportation of irradiated fuel, are made by truck. Irradiated fuel is brought a short distance by truck to the rail head, from which it is transported by rail to the reprocessing plant. The assumed quantities of materials and transportation distances for each shipment are presented in Table 2-2 from data given in References 5 and 7. Note that the conversion of enriched UF_6 to UO_2 powder and the subsequent steps in fuel fabrication are considered, for the purposes of transportation analysis, to be conducted at separate locations. The same assumption is made regarding the production of mixed oxide fuel rods and the fabrication of fuel assemblies containing mixed oxide fuel.

TABLE 2-2

TRANSPORTATION DATA FOR THE FUEL CYCLE

Transportation Step	Material Transported	Quantity per Shipment (MT)	Miles per Trip
Mine-Mill	Uranium Ore	27.2	5
Mill-UF_6 Conversion	U_3O_8	15.2	1000
UF_6 Conversion-Enrichment	Natural UF_6	12.7	500
Enrichment-UO_2 Plant	Enriched UF_6	11.0	750
UO_2 Plant-Fuel Fabrication	Enriched UO_2	4.5	750
Fuel Fabrication-LWR	Unirradiated Fuel	5.8	1000
LWR-Fuel Reprocessing	Irradiated Fuel	3.7*	Truck-20 Rail-1000
Recycle UF_6-Enrichment	Depleted UF_6	11.0	1000
PuO_2-Storage	PuO_2	0.26**	300
Natural UO_2-MO_x Plant	Natural UO_2	15.2	200
Stored PuO_2-MO_x Plant	PuO_2	0.26**	300
MO_x Fuel Rods-Fuel Fabrication	MO_x Fuel Rods	5.8	200

---

* 4.6 MT for no recycle case

** 0.32 MT for no recycle case

2.2 Normalization to the Annual Fuel Requirement of a Generic LWR

The fuel inventories in the initial core,[7] annual reloads,[5] and spent discharges[5] for the generic LWR are given in Table 2-3. Inventories are given for the case of no plutonium recycle as well as that of recycle. Both cases assume complete recycle of the uranium recovered from the spent fuel. The fuel management scheme assumed for the recycle case is the so-called 1.15 SGR model,[5] which recycles plutonium equivalent to 115% of that which could be self-generated within the reactor. The generic LWR is based upon a 1000 MWe boiling water reactor (BWR) design with an annual reload of about 172 fuel assemblies, or approximately 25% of the core.

Using the data given in Table 2-3, the flow of materials in the fuel cycle normalized to the annual operation of the generic LWR is developed for both the recycle and no recycle cases. The results, given in Table 2-4, are based upon attributing 1/30 of the mass flows to the initial core and 29/30 of the mass flows to the annual reloads (30-year reactor lifetime).[7]

Combining the information in Tables 2-1 and 2-4 gives the number of 1000 MWe LWR's serviced annually by each generic facility in the LWR supporting fuel cycle. The results, presented in Table 2-5, are used in Section 5 to normalize the estimated risks from each component of the fuel cycle to the annual operation of a generic LWR.

Similarly, the transportation data given in Table 2-2 are combined with the annual mass flows in Table 2-4 to provide estimates of the number

TABLE 2-3

FUEL INVENTORIES FOR GENERIC LWR

Initial Core[7]

Uranium (MT)	134
^{235}U Enrichment (%)	2.6

Annual Reload[5]

	Pu Recycle		No Pu Recycle
	U Rods	MO_x Rods*	
Uranium (MT)	19.0	12.7	32.3
^{235}U Enrichment (%)	2.6	0.71	2.6
Plutonium (MT)**	-	0.59	-

Annual Discharge[5]

	Pu Recycle	No Pu Recycle
Uranium (MT)	30.6	31.0
^{235}U Enrichment (%)	0.63	0.82
Plutonium (MT)	0.57	0.28

* MO_x rods, enriched to ~3.3% fissile, comprise 40% of the fuel rods

** 57.5% fissile plutonium

TABLE 2-4

ANNUAL MASS FLOWS IN THE LWR SUPPORTING FUEL CYCLE

Material	Annual Requirements	
	Pu Recycle	No Pu Recycle
Uranium Ore (MT)	60,200	82,900
U_3O_8 (MT)	120	166
Natural UF_6 (MTU)	89.9	141
Recycle UF_6 (MTU)	29.3	29.7
Separative Work (MTSWU)	69.5	102
Uranium Fuel Rods (MTU)	22.8	35.7
Spent Fuel (MTHM)	34.5	34.6
Pu to Storage ($MTPu_f$)	0.07	0.20
MO_x Fuel Rods (MTHM)	12.8	-

Assuming:

1% losses in reprocessing and fabrication

Enrichment tails assay = 0.25%

Ore contains 0.2% U_3O_8

TABLE 2-5

NUMBER OF 1000 MWe LWR's SERVICED ANNUALLY BY EACH GENERIC FACILITY IN THE LWR SUPPORTING FUEL CYCLE

Facility	Number of 1000 MWE LWR's Serviced Annually	
	Pu Recycle	No Pu Recycle
Mine	8.0	5.8
Mill	8.0	5.8
UF_6 Conversion Facility	42	29
Enrichment Plant	126	86
Uranium Fuel Fabrication Plant	25	25
Reprocessing Facility	43	43
Mixed Oxide Fabrication Plant	23	-
Pu Storage Facility	570	200

of shipments and resulting travel miles for each transportation step in support of the annual operation of a 1000 MWe LWR. These results, given in Table 2-6, are used as required in Section 5 to normalize the estimated risks from transportation to the annual operation of a generic LWR.

TABLE 2-6

NUMBER OF SHIPMENTS AND TOTAL TRAVEL MILES IN EACH FUEL CYCLE TRANSPORTATION STEP NORMALIZED TO THE ANNUAL OPERATION OF A 1000 MWe LWR

Transportation Step		Pu Recycle			No Pu Recycle	
		Number of Shipments	Total Travel Miles		Number of Shipments	Total Travel Miles
Mine-Mill		2,210	11,000		3,050	15,300
Mill-UF_6 Conversion		7.9	7,900		10.9	10,900
UF_6 Conversion-Enrichment		10.5	5,300		16.4	8,200
Enrichment-UO_2 Plant		3.1	2,300		4.8	3,600
UO_2 Plant-Fuel Fabrication		5.7	4,300		9.0	6,800
Fuel Fabrication-LWR		7.0	7,000		7.0	7,000
LWR-Fuel Reprocessing	Truck	9.3	190	Truck	8.5	170
	Rail	9.3	9,300	Rail	8.5	8,500
Recycle UF_6-Enrichment		3.9	3,900		4.0	4,000
PuO_2-Storage		0.27	80		0.63	190
Natural UO_2-MO_x Plant		0.91	180		-	-
Stored PuO_2-MO_x Plant		2.5	750		-	-
MO_x Fuel Rods-Fuel Fabrication		2.5	500		-	-

3. GENERIC MODELS FOR DEMOGRAPHY, DISPERSION, DOSE CONVERSION AND HEALTH EFFECTS

3.1 Demography

The distribution of human receptors in the vicinity of the generic fuel cycle facilities considered in this study is assumed to be uniform. A tabulation of these uniform population densities is given in Table 3-1. With the exception of mixed oxide fuel fabrication, plutonium storage, and transportation, the tabulated numbers are the averages for the population density ranges estimated in Reference 7. The population density in the vicinity of the mixed oxide fabrication plant is taken from estimates given in Reference 5, and the plutonium storage population density is assumed to be the same as that of mixed oxide fuel fabrication.

The population density for transportation is obtained by integrating over the population density distributions given in Reference 17 for highways and railroads.

Most of the fuel cycle facilities considered in this study are surrounded by areas of land which are restricted from use by the general public. This "restricted area" is tabulated for each of the generic fuel cycle facilities in Table 3-2, accompanied by the closest distance to the unrestricted area, obtained by assuming a circular plot. With the exception of the numbers for mixed oxide fuel fabrication and plutonium storage, which are obtained from Reference 5, these data are obtained from Reference 7. Although the distance to the unrestricted area may be ignored in estimating population doses from airborne radiological

TABLE 3-1

POPULATION DENSITIES IN THE VICINITY OF THE GENERIC FUEL CYCLE FACILITIES

Facility	Population Density (persons/mi^2)
Mine	7.5
Mill	7.5
UF_6 Conversion Plant	47.5
Enrichment Plant	35
Uranium Fuel Fabrication	240
Reprocessing	90
Mixed Oxide Fuel Fabrication	100
Plutonium Storage	100
Transportation	290

TABLE 3-2

AVERAGE DISTANCES TO UNRESTRICTED AREA FOR GENERIC FUEL CYCLE FACILITIES

Facility	Restricted Area (acres)	Approximate Distance to Unrestricted Area (meters)
Mine	3000	1000
Mill	(co-located with mine)	1000
UF_6 Conversion Plant	1400	1000
Enrichment Plant	1500	1250
Uranium Fuel Fabrication	500	750
Reprocessing	2000	1500
Mixed Oxide Fuel Fabrication	1000	1000
Plutonium Storage	250	500
Transportation	0	10*

---

* Assumed distance to populated area.

effluents, it is important in assessing the population dose from the prompt gamma and neutron radiation resulting from a postulated criticality incident (see Section 3-2).

For the estimation of population dose from liquid effluents, the approach outlined in Reference 11 is adopted. It is assumed that 2000 persons/km of a generic river for 300 km downstream from the point of release drink water from the river. This places the population at risk at approximately 6 x 10^5 individuals. For mines and mills, the population at risk is taken to be approximately 44,000 individuals, obtained by reducing the population density by a factor of 0.037 and multiplying the result by a factor of two, in order to account for the clustering of people around water supplies in arid regions.

3.2 Dispersion and Dose Conversion

All doses estimated in this report were calculated to represent a 50-year dose commitment* from a given accident release (the routine operational release doses given for comparison were calculated for a single year's release). The estimates incorporate long-term persistence in the environment as well as residence of the material in the body after uptake has stopped. In general, the dose conversion calculations have been made following the recommendations of the International Commission on Radiological Protection (ICRP). Specifically, internal dose conversion calculations have been made using the maximum permissible concentrations given by ICRP Committee 2,[20] as updated in ICRP publications 6[21] and 10.[22] For isotopes appearing in the accident source terms which are not given in ICRP publications, dose conversion factors were taken from the latest Oak Ridge data as contained in the INREM[23] code. For the few short-lived isotopes which are neither in ICRP nor INREM listings, dose conversion factors were derived using half-life and decay energy data from the Radiological Health Handbook,[24] along with biological data from ICRP II on isotopes of the same element. Isotopes with radiological half-lives of less than 10 minutes were neglected, as their contribution to the 50-year population dose commitment is negligible.

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* This time period includes the bulk of the dose delivered by most of the isotopes released in the fuel cycle. For the very long-lived radionuclides, such as plutonium-239 and iodine-129, much longer time periods are required. However, quantitative estimates of dose over thousands to millions of years are highly uncertain. Earlier studies have arbitrarily chosen 100 years[18] and 70 years[19] as cutoffs. Considering the scoping nature of this study, the use of a 100, rather than a 50-year integration period for estimating dose commitment would not significantly affect the results.

Dose conversion factors for airborne insoluble particulates followed the approach of EPA,[11] converting from the 1959 lung model of ICRP II to the newer lung model given in the ICRP Task Force report, "Deposition and Retention Models for Internal Dosimetry of the Human Respiratory Tract."[25] Typically, this provides an increase in dose for a given airborne concentration by a factor of eight. Dose conversion calculations for airborne radon-222 were made using the EPA suggested value of 1 pCi/m^3 = 4 mrem/year,[11] rather than the 1.5 mrem/year value of ICRP II.

Plume submersion doses were calculated using a semi-infinite cloud model. Total body and organ doses were calculated on the basis of total energy deposited with 5 cm of tissue shielding. Lung doses include both the external, 5 cm dose and dose from inhaled material, following the treatment of Snyder.[26] Ground plane irradiation dose calculations were made using Oak Ridge EXREM III[23] dose for gamma irradiation 100 cm above an infinite ground plane.

All dose calculations included resuspension of deposited materials and build-up of daughter products during the 50 years following the initial release. The resuspension coefficient was taken as 10^{-5} m^{-1} initially, and was assumed to decay to 10^{-9} m^{-1} with a 50 day half-life. For long-lived isotopes, this model leads to a resuspension inhalation dose contribution equivalent to 64% of initial inhalation.

Dietary doses were calculated from intake of vegetation, milk and meat. The assumed intake of each type of food used in the calculation was:

Vegetation - 400 grams/day
Milk - 350 grams/day
Meat - 250 grams/day

Isotopes were assumed to be deposited directly on vegetation as well as deposited on soil and taken up by plant roots. Transfers of isotopes from soil to plants and from plants to animals were based on relative concentration data of stable elements in the pathway of concern as given by Livermore and Oak Ridge data.[27-29]

Since this study deals with a number of different existing and planned facilities, the dispersion modeling was done on a generic rather than site-specific basis. For generic facilities, the actual spatially-dependent population densities and dispersion factors are not known, but reasonable average population doses can be determined assuming uniform, average population densities. In general the total population dose is given by:

$$\text{Population dose} = K \int_{\text{space}} \Psi_{\vec{r}} \, P_{\vec{r}} \, d\vec{r} \, ,$$

where $\Psi_{\vec{r}}$ is the pointwise concentration time integral and $P_{\vec{r}}$ is the pointwise population density. For a general population distribution the spatial integral will be a function of both $\Psi_{\vec{r}}$ and $P_{\vec{r}}$, but for a uniform population density the population dose may be written as:

$$\text{Population dose} = K \bar{\Psi} P \, ,$$

where $\bar{\Psi}$ is the spatially averaged concentration time integral appropriate for the population of P individuals. At any point, the concentration time integral, $\Psi_{\vec{r}}$, will be related to the ground concentration, $W_{\vec{r}}$, and the deposition velocity, Vg, by:

$$V_g = W_{\vec{r}} / \Psi_{\vec{r}} \, .$$

Thus, the population dose can be expressed as:

$$\text{Population dose} = K \frac{\bar{W} P}{V_g} \, ,$$

where $\overline{W}$ is the average ground concentration. In the above equation only the average ground concentration, $\overline{W}$, is needed. Noting that whatever is released will eventually settle, we can define the average $\overline{W}$ over a large arbitrary areas as:

$$\overline{W} = Q/A \,,$$

where Q is the total source released. This gives:

$$\text{Population Dose} = K \cdot \frac{Q}{V_g} \cdot \frac{P}{A} \,,$$

where P/A is the average population density (people per square meter),* Q is the total source released (Curies), Vg is the deposition velocity (meters per second) and K is the dose conversion factor (rem per Ci-sec/m^3).

The above equation was used to determine population doses for all particulate emissions using the average population densities given in Table 3-1. A deposition velocity of 0.01 m/sec was used in all the calculations.

The above equation does not take into account radioactive decay during dispersion. Assuming mixing in a plume limited to a height, L, it can be shown that the population dose is given now by the formula:[30]

$$\frac{K \; Q}{(V_g + \lambda L)} \frac{P}{A} \,,$$

where $V + \lambda L$ is the effective plume depletion rate per unit height. The mixing height boundary, L, was taken as 1000 meters. For short-lived radioactive gases, the same formula as above was used with $V_g = 0$. For long-lived gases, namely H-3, C-14 and Kr-85, dilution of the gases into the earth's reservoirs and exposure of the total world's population was used to calculate doses. Kr-85 was diluted in the world's atmosphere (3.8×10^{18} m^3) and doses calculated for exposure of 4×10^9 people. Tritium was diluted in the circulating water volume of 7×10^{15} m^3, and C-14 releases were assumed to reach isotopic equilibrium with the CO_2 in the earth's atmosphere.

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* It should be noted that a time-invariant population density has been assumed in this study. In actuality, the population will be increasing over the time period used in calculating dose commitment, leading to somewhat higher estimates. Considering the other approximations employed, however, the neglect of this effect does not significantly affect the results.

For releases to waterways, the EPA water pathway model[11] was used to calculate doses. Basically this model assumes a river with a flow of about 1000 cfs, with a density of people of 2000 persons/km for 300 km downstream who drink water from the river. The effective χ/Q for this river model is 4×10^{-7} $\frac{pCi/l}{pCi/sec}$, when all dilution factors are taken into account.

For mines and mills, which are located in relatively remote parts of the country, special models were used in calculating doses. For waterborne effluents, the EPA model for mills was used which takes a river flow of 1/20 of that given above, and the population at risk was taken as 44,000, as opposed to the 600,000 used for the other facilities. For radon releases to air from the mines and mills, the effective travel time before decay is sufficiently large that populations much beyond the mill proper may be exposed. Thus, three different generic population densities were used in assessing dose. Within 50 miles of the facility, the generic population density of 7.5 people/mile[5] given in Table 3-1 was used; for distances between 50 and 500 miles a population density of 50 people/mile,[5] typical of the Western United States, was used; and for distances from 500 to 2000 miles a population density of 160 people/mile,[5] typical of the Eastern United States, was used. Interestingly, the largest dose commitment from radon releases, using this model, is that received by the far high density region corresponding to the Eastern United States. Note also that when build-up of Pb-210 from the decay of Rn-222 is taken into account, uptake of lead in the diet (which persists over 50 years) yields a larger dose to man than from direct inhalation of the Rn-222 in the initial plume passage.

The final population dose commitment values used in the calculations are given in Appendix A. It is worth noting here that the organ receiving the largest dose is selected in each case as the critical organ, regardless of the actual chemical characteristics of the material at the time of release. This approach, which provides a conservative estimate of dose commitment, is necessitated by uncertainties in the long-term behavior of these materials in the environment.

For criticality incidents, the population dose from the prompt gamma and neutron radiation emitted by the fission burst must be added to the dose from radiological effluents released to the atmosphere. The dose received by individuals from the direct radiation "shine" of a 10^{18} fission burst is given as a function of distance from the burst in Table 3-3. The dose at 100 meters was obtained from the value given in Reference 8. However, in addition to the geometrical inverse square attenuation, the dose was further attenuated by an air removal cross section of 0.027 cm^{-1} for both neutrons and gamma rays, assuming dry air. It is also important to note that the dose at 100 meters has been corrected for facility shielding equivalent to 8 inches of ordinary concrete (fractional transmission of 0.17 for the neutron radiation and 0.22 for gamma radiation).

TABLE 3-3

RADIATION DOSE TO AN INDIVIDUAL AS A FUNCTION OF DISTANCE RESULTING FROM THE DIRECT "SHINE" OF A CRITICALITY INCIDENT (10^{18} FISSIONS)

Distance (meters)	Radiation Dose (rem)
100	1.6
500	1.7×10^{-2}
600	8.4×10^{-3}
700	4.6×10^{-3}
800	2.4×10^{-3}
900	1.3×10^{-3}
1000	5.9×10^{-4}
1250	2.2×10^{-4}
1500	6.8×10^{-5}
2000	7.6×10^{-6}
3000	1.2×10^{-7}
4000	2.5×10^{-9}
5000	5.9×10^{-11}
10000	7.5×10^{-19}

3.3 Health Effects

Most of the radiation exposures to the general public estimated by this study are well below the range in which effects have been directly observed. Accordingly, theoretical estimates of the relationship between absorbed dose and health effects must be relied upon to provide quantitative estimates. There exists considerable disagreement within the scientific community as to the appropriate relationship for such estimates. However, most advisory and standards setting groups suggest that it is correct and prudent to use the linear, non-threshold hypothesis in standard setting. According to this hypothesis, there is a linear relationship between the total accumulated dose and the number of effects (for all types of effects observed) from zero exposure to the highest exposure which does not cause acute mortality before expression of the delayed effect. Using this hypothesis, if 1 rem produces 10 effects, then 2 rems produce 20 effects and 0.1 rem produces 1 effect. Differences in magnitude of dose or dose rate do not have any influence on the relationship if the linear, non-threshold hypothesis is adopted.

Because of the widespread use of this model in standards setting, and since no other model has achieved widespread acceptance for quantitative estimates of health effects at low levels of exposure, the linear, non-threshold dose-response relationship is adopted for use in this study. Table 3-4, developed by EPA, lists the number and types of effects expected after the exposure of a population of one million to a total of one million man-rem.[12] This table was developed by integrating forward over time the estimates of annual effects from the BEIR report.[13]

TABLE 3-4

EFFECTS OF A SINGLE EXPOSURE OF A POPULATION OF ONE MILLION TO ONE REM PER PERSON (10^6 MAN-REM)[12]

Target Organ	Somatic Effects (excess cancers)	Genetic Effects (congenital defects)
Total Body	200 fatal 200 non-fatal	300
Distributed as follows:		
Breast	90	---
Lung	40	---
GI Tract	62	---
Bone	12	---
Leukemia	49	---
Other	147	---
Thyroid	62	---
	400	

All of the effects to populations of exposures at low dose levels of radiation are delayed. These delayed effects show up after a relatively long latency period and fall into two broad categories - somatic and genetic. Somatic effects are generally some form of cancer, and are the effects which are quantified in this study. Depending on the distribution of exposure in the body, these cancers develop at specific locations in the body or in specific organ systems. Approximately 50% of the cancers are fatal. Delayed genetic effects apparently arise from mutations in the sperm and ova, and usually result in children with hereditary congenital defects.

In this study, Table 3-4 is used by determining the population dose to the "critical organ" (the body organ receiving the maximum absorbed dose), and converting this dose to health effects using the conversion factor in the table. For example, if the population dose to the GI tract were x man-rem, the number of estimated health effects would be given by $62x/10^6$. Since estimated effects are assumed to be cumulative across the total body and tissues, the estimated effects to the remainder of the total body must be added to the effects to the critical organ. For example, if the corresponding dose to the total body in the above example were y man-rem, the number of additional health effects would be given by $(400-62)y/10^6$. The total estimated health effects, then, would be given by the sum of the effects to the critical organ and the total body.

For a few of the postulated accidents, acute effects as well as long-term delayed effects would be expected. The number of acute effects are expected to occur within a period of two to three months after the exposure. For purposes of this study, only possible deaths were considered, and a very simple relation was used to estimate the number of fatalities. Namely, for exposures in the range of 200 to 600 rem, 50% of the exposed population is assumed to receive a lethal dose, and above 600 rem, 100% of the exposed population is assumed to die. The population dose commitment to the surviving individuals in the 200 to 600 rem range, and to those individuals receiving less than 200 rem, is added to the low-level dose commitment for the purposes of estimating long-term somatic health risks.

4. SOURCE TERMS AND LIKELIHOODS

For the purpose of ease in documentation and subsequent expansion or revision, the accidents examined in this study are organized within the broad categories given in Table 4-1. Containment failures, Category C, include vessel, pipe, valve, and ventilation system failures of all types. A loss-of-coolant incident, for example, would be incorporated within this category. External events, including acts-of-God and inadvertent human-induced events, would comprise a separate category were they included within the scope of this study.

The source terms and likelihoods estimated for each postulated accident are generally synthesized from the existing literature. The major sources of information are generic environmental statements and/or safety assessments.

These documents generally derived their data from individual detailed environmental statements, safety analysis reports, incidents abstracts, or compliance files. As discussed in Section 1.3, the original data are subject to considerable manipulation in this study. This includes consolidation, renormalization, and reinterpretation. Little attempt has been made in this preliminary risk assessment to utilize a consistent methodology for deriving source terms or likelihoods.

Accident likelihoods have been the most difficult data to obtain and are generally the softest numbers quoted in this study. A cursory survey was made of incidents on record in government[31] and commercial[32] facilities to augment the information discussed above. The accidents

TABLE 4-1

CATEGORIES OF ACCIDENTS

A. Explosions
B. Fires
C. Containment Failures
D. Criticality
E. Retention Pond Releases

identified in this survey* which are pertinent to this study are tabulated in Tables 4-2 through 4-5. Note that, in general, the magnitude of the release has not been documented. Criticality incidents are well documented in a number of sources,[31,33,34] and are therefore not tabulated here.

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* This survey was by no means comprehensive.

TABLE 4-2

EXPLOSION INCIDENTS IN FUEL CYCLE FACILITIES

Info. Source	Date	Where	Process Type	Release Magn.	Details
NSIC	1/67	NUMEC	MO_x Fab.	0.1 gm Pu	Glovebox breach from projectiles resulting from H_2O_2 decomp.
NSIC	12/72	GUNF, Elmsf.	MO_x Fab.	?	Glovebox containing sintered $U\text{-}PuO_2$
NSIC	12/68	UNC, Wood River	U Fab.	?	Explosion in scrap recovery
WASH-1192	7/59	Hanford	U Fab.	Probably zero	Autoclave explosion
WASH-1192	1/60	ORNL	U Fab.	?	Sintering furnace explosion
WASH-1192	4/60	Hanford	Repro.	?	Dissolver explosion
WASH-1192	12/60	Hanford	U. Fab.	Probably zero	Autoclave explosion
WASH-1192	8/65	Battelle-Northwest	MO_x Fab.	?	Cleaning fluid in glovebox ignited
WASH-1192	4/68	Mound Lab.	MO_x Fab.	?	Glovebox drying oven explosion

TABLE 4-3
FIRE INCIDENTS IN FUEL CYCLE FACILITIES

Info. Source	Date	Where	Process Type	Release Magn.	Details
NSIC	1/74	NUMEC	U Fab.	?	Hydrogen ignited, burned filter
NSIC	9/72	UNC	U Fab.	?	Fire in scrap recovery burned through stack
NSIC	4/71	UNC-Hematite	U Fab.	?	Leak in off-gas system of UF_6-UO_2 converter. Hydrogen release ignited
NSIC	3/72	NFS, Erwin	U Fab.	?	Flash fire in tray dissolver
NSIC	12/68	Petrotomics	Mill	?	Fire in solvent extraction circuit
NSIC	7/63	Rocky Flats	MO_x Fab.	?	Ion-exchange recovery fire and explosion
NSIC	'62-'68	Several	General-Ion Exchange	?	Seven ion-exchange resin fires from self-ignition of nitrate-form resins
NSIC	3/66	B&W	U Fab.	?	Dissolver fire
WASH-1192	11/56	Paducah	Enrich.	?	Fire spread thru roof ($2,100,000)
WASH-1192	12/62	Paducah	Enrich.	?	Major fire in gaseous diffusion cell ($2,900,000)
WASH-1192	10/64	Savannah River	Repro.	?	Fire occurred around anion exchange column in hot canyon

TABLE 4-4

LOSS OF CONTAINMENT INCIDENTS IN FUEL CYCLE FACILITIES

Info. Source	Date	Where	Process Type	Release Magn.	Details
NSIC	1/73	NUMEC-Apollo	U Fab.	2.5gms ^{235}U to	Extraction column rupture
NSIC	4/71	NFS-West Valley	Repro.	?	Valve left open in Pu Prod. Storage Tank
NSIC	4/69	B&W	MO_xFab.	?	Leak in Pu calcining furnace
NSIC	3/69	NUMEC	MO_xFab.	?	Leak in bag containing glovebox sweepings
NSIC	7/68	Kerr-McGee-Okl.	Fab.	100 lbs. UF_6	Inadvertent valve open on vaporiz. equipment
NSIC	5/68	NUMEC, Apollo	U Fab.	?	Leaking gasket on blender
NSIC	11/67	NUMEC, Apollo	MO_x Fab.	?	Pump leak
NSIC	12/68	Allied Chem., Metr.	UF_6 Conv.	90 lbs. UF_6	Ruptured valve in dist. process
NSIC	11/67	NUMEC, Apollo	U Fab.	600gms ^{235}U to River	Inadvertently opened valve to dump tank

TABLE 4-4
(continued)

Info. Source	Date	Where	Process Type	Release Magn.	Details
NSIC	2/69	NFS	U Fab.	?	5 Kg UO_2 spilled when blender discharge valve opened accidentally
NSIC	5/72	NFS, W. Valley	Repro.	?	Acid and water released into extraction aisle
NSIC	6/71	NFS, W. Valley	Repro.	?	Release of contaminated water in sewage
WASH-1192	8/59	Savannah River	Repro.	?	Leakage from waste evaporator
WASH-1192	9/60	Savannah River	Repro.	?	Contamin. cooling water discharged from canyon onto floor
WASH-1192	11/60	ORNL	Enrich.	3077 kg U	Ten-ton UF_6 cylinder rupture
WASH-1192	11/60	Natl. Lead	U Fab.	?	Sl. enr. UF_4 lost thru stack of dust collector
WASH-1192	5/61	Mound Lab.	U Fab.	?	Caustic scrubber radiation release
WASH-1192	11/61	Hanford	U Fab.	1089 lbs. depl U to sewer	?
WASH-1192	2/66	Natl. Lead	U Fab.	3844 Lbs. UF_6 (to environ.?)	Operator inadvertently unscrewed valve from head of a 10 ton cylinder

TABLE 4-4
(continued)

Info. Source	Date	Where	Process Type	Release Magn.	Details
WASH-1192	11/69	Savannah River	Repro.	?	Acidic waste soln. inadvertently transf. to underground waste system due to leak. valve, and lost
Health Physics Vol. 11, pp. 1009-1015	9/63	Hanford	Repro.	60 Ci I-131	?

TABLE 4-5

FILTER FAILURE INCIDENTS IN FUEL CYCLE FACILITIES

Info. Source	Date	Where	Process Type	Release Magn.	Details
NSIC	1/74	NFS-Erwin	U Fab.	?	Filter in vacuum cleaner failed
NSIC	7/73	GE,N.C.	U Fab.	?	Roughing filter plugged
NSIC	1/73	UNC,New Haven	U Fab.	?	Holes in HEPA from calciner
NSIC	6/70	NUMEC, Apollo	MO_x Fab.	67 Ci Pu to air	Dissolver exhaust line filter failed
NSIC	3/68	NFS	U Fab.	Monthly allowance (?)	Filter failure
NSIC	9/68	NFS, W. Valley	Repro.	11% monthly allow.	Filter failure
NSIC	8/67	UNC-Wood River	U Fab.	1 gm U-235	Ruptured exhaust filter
NSIC	11/72	UNC-New Haven	U Fab.	?	Acid fumes caused failure on non-acid resistant HEPA filter

4.1 Mining

No accidents have been identified in the mining of uranium ore which would result in significantly higher releases to the environment than incurred during normal operations. Fire or earth collapse could lead to occupational injuries, but would not result in airborne radioactive effluents (uranium-bearing dusts and radon and its daughters) in excess of those released when the ore body is exposed and broken up during mining operations. Power failure to the ventilation system in an underground mine could lead to a buildup of radon, but the integrated release over time would be unchanged.

A failure of the mine drainage system to dewater the mine area could lead to flooding, or flooding of the mine area could result from natural causes. However, the total release of activity in mine drainage water would not be substantially higher than that released during normal mine drainage.

At some mines, the drainage water is held in retention before being released. Failure of a retention pond dike could result in the release to the environment of the contaminated drainage water together with suspended solids from the mines. Since the fate of the suspended solids is site-specific, and the activity in the mine drainage water is generally not significantly greater than that of natural mineral springs in the vicinity of the mine, this accident was not analyzed in the current study.

4.2 Milling

The accidents considered in uranium milling facilities, keyed to the accident categories given in Table 4-1, are listed in Table 4-6.

Other incidents may be postulated or have occurred in mills from which the environmental risks are deemed insignificant from the existing information. These include local fires, overflows from process tanks, failures of process lines, failures of offgas filtration or scrubbing systems, or storage tank spills.[7]

Natural or man-induced disasters, such as tornados, earthquakes, floods, or missile impacts are highly unlikely, although their occurrence could result in varying releases of activity. The risks associated with these events were not analyzed in the current study.

Table 4-7 provides historical data on uranium mills currently in operation.[35,36] Historical data on mills no longer in operation are provided in Table 4-8.[37]

B.1 Fire in Solvent Extraction Circuit

In the solvent-extraction step of the milling process, the uranium is purified and concentrated. This is accomplished by contacting the gaseous phase from the leaching step, which contains the uranium and impurities, with an organic solvent. The solvent extraction circuit is likely to be in a separate building containing several thousand gallons of solvent (mostly kerosene) and several thousand pounds of natural

TABLE 4-6

MILLING ACCIDENTS

B.1	Fire in solvent extraction circuit
E.1	Release of tailings slurry from tailing pond
E.2	Release of tailing slurry from tailings distribution pipelines

TABLE 4-7

URANIUM MILLS IN OPERATION AS OF MARCH, 1975[35,36]

COMPANY	LOCATION	YEAR OPERATIONS INITIATED	NOMINAL CAPACITY (Tons of Ore/Day)
Anaconda Company	Grants, New Mexico	1953	3000
Atlas Corporation	Moab, Utah	1956	800-1500
Conoco & Pioneer Nuclear, Inc.	Falls City, Texas	1961	220-1750
Cotter Corporation	Canon, City, Colorado	1958	150-450
Dawn Mining Company	Ford, Washington	1957	0-400
Exxon, U.S.A.	Powder River Basin, Wyoming	~1971	2000
Federal-American Partners	Gas Hills, Wyoming	1959	500-950
Kerr-McGee Nuclear	Grants, New Mexico	1958	3600-7000
Petrotomics Company	Shirley Basin, Wyoming	1962	525-1500
Rio Algom Corp.	La Sal, Utah	~1972	500
Union Carbide Corp.	Uravan, Colorado	~1950	0-1300
Union Carbide Corp.	Natrona County, Wyoming	1960	1000

TABLE 4-7
(continued)

COMPANY	LOCATION	YEAR OPERATIONS INITIATED	NOMINAL CAPACITY (Tons of Ore/Day)
United Nuclear-Homestake Partners	Grants, New Mexico	1958	1650-3500
Utah International, Inc.	Gas Hills, Wyoming	1958	750-1200
Utah International, Inc.	Shirley Basin, Wyoming	1971	1200
Western Nuclear,Inc.	Jeffrey City, Wyoming	1957	400-1200
TVA (Mines Development, Inc.)	Edgemont, South Dakota	1956	250-500

TABLE 4-8

URANIUM MILLS NO LONGER IN OPERATION[37]

COMPANY	LOCATION	YEARS OPERATED	TONS OF TAILINGS
Foote Mineral Co.	Monument, Arizona	1955-1967	1,200,000
El Paso Natural Gas	Tuba City, Arizona	1956-1966	800,000
Vanadium Corp.	Durango, Colorado	1943-1963	1,555,000
Climax Uranium Co.	Grand Junction, Colorado	1951-1970	1,900,000
Colorado Ventures	Gunnison, Colorado	1958-1962	540,000
Union Carbide Corp.	Maybell, Colorado	1957-1964	2,600,000
Vanadium Corp.	Naturita, Colorado	1939-1963	704,000
Union Carbide Corp.	New Rifle, Colorado	1958-1972	2,700,000
Union Carbide Corp.	Old Rifle, Colorado	1924-1958	350,000
North Continent	Slick Rock, Colorado	1931-1943	37,000
Union Carbide	Slick Rock, Colorado	1957-1961	350,000
Michigan Chemical Corp.	Lowman, Idaho	1955-1960	90,000
United Nuclear Corp.	Ambrosia Lake, New Mexico	1958-1963	2,600,000
Foote Mineral Co.	Shiprock, New Mexico	1954-1968	1,500,000
Atlantic Richfield	Lakeview, Oregon	1958-1960	130,000
Susquehanna Western	Ray Point, Texas	1970-1973	490,000
Union Carbide Corp.	Green River, Utah	1958-1961	123,000
A-Z Minerals	Mexican Hat, Utah	1957-1965	2,200,000
Vitro Corp.	Salt Lake City, Utah	1951-1968	1,700,000
Western Nuclear	Converse County, Wyoming	1962-1965	187,000

uranium.[7] The flammability of the solvent provides the potential for a serious fire.

Source Term

Two to three thousand pounds of uranium were present in the solvent extraction circuits involved in major fires in 1968.[7] This is a reasonable inventory for a 960 MT U_3O_8/yr mill which produces approximately 6000 lbs/day of uranium.

If 1% of the uranium were dispersed, as assumed in estimates of the amount of plutonium dispersed in a solvent extraction fire,[9] 20 to 30 pounds of uranium could be released to the atmosphere. If 0.5-0.7% were dispersed, as assumed in estimates of the amount of plutonium released from a fire near nitrate blending tanks,[8*] 10 to 20 pounds of uranium could be released to the atmosphere.

It is assumed that 99% of the radium had been removed in the leaching step, but that the thorium in equilibrium with uranium is present in the solvent extraction step. Thus the estimated concentrations of radioisotopes in the material released are: 3.3×10^{-7} Ci/gm of ^{238}U, 1.54×10^{-8} Ci/gm of ^{235}U, 3.52×10^{-7} Ci/gm of ^{234}U, 3.31×10^{-7} Ci/gm of ^{234}Th, 3.31×10^{-7} Ci/gm of ^{230}Th, and 3.31×10^{-9} Ci/gm of ^{226}Ra.

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[*] Experiments have demonstrated that approximately 80% of the uranium aerosol generated from a nitrate solution involved in a gasoline fire is the respirable size range.[8]

Likelihood

From chemical industry data, the probability of major fires per plant-year is estimated to be 4×10^{-4}.[8] However, at least two major solvent extraction circuit fires are documented in the literature.[7] From the data in Tables 4-7 and 4-8, there have been 515 plant-years of mill operation, or the equivalent of 282 plant-years handling 480,000 MT ore/yr. Thus, from the historical incidents, the likelihood of a major solvent extraction fire is in the range of 3 to 7×10^{-3}/plant-year.

Estimate: *Source term* $\approx 3.3 \times 10^{-3}$ *Ci* ^{238}U, 1.5×10^{-4} *Ci* ^{235}U, 3.5×10^{-3} *Ci* ^{234}U, 3.3×10^{-3} *Ci* ^{234}Th, 3.3×10^{-3} *Ci* ^{230}Th, 3.3×10^{-5} *Ci* ^{226}Ra *to air. Likelihood* $\approx$ 3×10^{-3} *to* 4×10^{-4}*/plant-year.*

E.1 Release of Tailings Slurry from Tailings Pond

The solid residues (tailings) from the leaching circuit of the milling process are suspended in a wash solution and pumped to the tailings retention pond. The aqueous phase from the solvent extraction circuit, which is called the raffinate and contains most of the impurities in the ore, is also pumped to the tailings retention pond. The tailings are composed mostly of sandstone and clay particles, and contain about 85% of the activity originally in the ore. The tailings pond is constructed by erecting an earth fill, clay core dam across a natural basin. The system may hold on the order of 1,500,000 MT of solid tailings.[7] Tailings dams have been known to fail due to flooding, earthquake, or inattention.

Source Term

Accidental tailings slurry releases have been documented and the resulting release estimates compiled.[7] Table 4-9 contains a summary of recorded incidents in the period from 1959 to 1971. From these data, the average releases from tailings dam failure or flooding are 3000 m^3 of liquids and 7.5×10^6 lbs. of solids.

The estimated concentrations of radioactive effluents in the waste liquor from a generic uranium mill are given in Table 4-10.[11] The estimated total specific activity of solids is 3.7 μCi/lb.[7] The solids, however, are assumed to deposit in the vicinity of the entry point to the watercourse.

Likelihood

Eight out of eleven of the releases documented in Table 4-9 reached the watercourse, and of the total of 12 recorded incidents, eight involved dam failure or flooding. Referring to Tables 4-7 and 4-8, there were 270 plant-years of operation in the period 1956 through 1970, or the equivalent of 153 plant-years handling 480,000 MT ore/yr. Using these historical data, the likelihood of release from the tailings pond itself to the watercourse is in the range of 2 to 4 $\times 10^{-2}$/plant-year.

TABLE 4-9

SUMMARY OF ACCIDENTAL TAILINGS SLURRY RELEASES[7]

Cause	Solids Released (lbs.)	Liquids Released (gallons)	Reached Watercourse
Flash flood	30×10^{6}	$3.3 \times 10^{6*}$	yes
Dam failure	$2 \times 10^{6*}$	2.4×10^{5}	yes
Dam failure	1×10^{6}	$1 \times 10^{5*}$	no
Dam failure	4×10^{5}	$5 \times 10^{4*}$	yes
Pipeline failure	6×10^{5}	6×10^{4}	yes
Flooding	No quantitative information		?
Pipeline failure	$1.4 \times 10^{5*}$	1.6×10^{4}	small amount
Flooding	$4 \times 10^{6*}$	4.4×10^{5}	yes
Dam failure	$3\text{-}30 \times 10^{6*}$	$3\text{-}30 \times 10^{5}$	yes
Pipeline failure	$3 \times 10^{5*}$	3.5×10^{4}	yes
Dam failure	$2 \times 10^{4*}$	2×10^{3}	no
Pipeline failure	No quantitative information		no

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* Assuming equal weights of solids and liquids released and density of liquids approximately 9 lbs/gallon.

TABLE 4-10

CONCENTRATIONS OF RADIOACTIVE EFFLUENTS IN WASTE LIQUOR FROM THE MODEL URANIUM MILL[11]

Contaminant	Concentration
Uranium - natural	8.0×10^{-7} μCi/ml
Radium 226	3.5×10^{-7} μCi/ml
Thorium 230	2.2×10^{-5} μCi/ml

It should be pointed out that these historical incident data may be misleading, since the evaluation of early dike construction led to a determination that dikes need strengthening. Mills having dikes similar in construction to those that failed were required to strengthen the dikes and new mills were required to use new construction standards.

__Estimate:__ *Source term* $\approx 1.1 \times 10^{-3}$ *Ci* ^{238}U, 5.3×10^{-5} *Ci* ^{235}U, 1.2×10^{-3} *Ci* ^{234}U, 1.1×10^{-3} *Ci* ^{234}Th, 1.1×10^{-3} *Ci* ^{226}Ra, 6.6×10^{-2} *Ci* ^{230}Th *to watercourse; Likelihood* $\lesssim 4 \times 10^{-2}$*/plant year.*

E.2 Release of Tailings Slurry from Tailings Distribution Pipelines

Tailings distribution pipelines have been known to fail, resulting in the accidental release of tailings slurry to the watercourse.

Source Term

From the data given in Table 4-9, the average release from incidents on record involving the failure of pipelines in the tailings distribution system is 130 m^3 of liquids and 3.5×10^5 lbs. of solids.

Likelihood

Of the total of 12 recorded incidents documented in Table 4-9, four involved pipeline failures. Using the same approach as that employed in deriving the likelihood of dam failures, the likelihood of release from tailings distribution pipelines is in the range of 1 to 2×10^{-2}/plant/yr.

Estimate: *Source term* $\approx 4.8 \times 10^{-5}$ *Ci* ^{238}U, 2.3×10^{-6} *Ci* ^{235}U, 5.2×10^{-5} *Ci* ^{234}U, 4.8×10^{-5} *Ci* ^{234}Th, 4.8×10^{-5} *Ci* ^{226}Ra, 2.9×10^{-3} *Ci*^{230}Th *to watercourse. Likelihood* $\approx 1 \times 10^{-2}$*/plant-year.*

4.3 UF_6 Conversion

The accidents considered in UF_6 Conversion facilities, keyed to the accident categories given in Table 4-1, are listed in Table 4-11.

Other incidents have occurred or are possible for which the associated environmental risks are judged to be insignificant in comparison with those of the accidents listed in Table 4-11. These include yellowcake spills at the head end of the process, local fires, loss of refrigeration to cold traps, or small releases of UF_6 from valve or process line failures.[7] There are also credible accidents which would release toxic chemicals to the environment, such as the rupture of a hydrogen fluoride tank or a leak in the fluorine production plant, but these are considered outside of the scope of the current study. Also, the risks associated with the release of radioactivity from disasters, such as tornados, or missile impact, were not analyzed.

There are currently two commercial facilities for the production of uranium hexafluoride, operated by Allied Chemical Corporation and Kerr-McGee Corporation, respectively. The former plant has operated a total of 14 years since 1959 at production capacities up to 5000 MTU/yr. The latter initiated operations in 1970 and has accumulated 5 to 6 years at production capacities close to 5000 MTU/yr.

A.1 Uranyl Nitrate Evaporator Explosion

In the wet chemical solvent extraction process for producing uranium hexafluoride, the uranium concentrate is dissolved in nitric acid and sent to

TABLE 4-11

UF_6 CONVERSION ACCIDENTS

A.1 Uranyl nitrate evaporator explosion.

A.2 Hydrogen explosion in the reduction step of the process.

B.1 Fire in the solvent extraction operation.

C.1 Release from a hot UF_6 cylinder.

C.2 Valve rupture in the distillation step.

E.1 Release of raffinate from the waste retention pond.

the uranium-extraction column, where the aqueous solution of uranyl nitrate is extracted countercurrently with tributyl phosphate in hexane. The uranium is then reextracted as uranyl nitrate solution into a large volume of water, which is concentrated by evaporation. An explosion could occur in the evaporator from a "red-oil" reaction. "Red-oil" is a material that is formed from a heavy metal nitrate and nitric acid solution mixed with tributyl phosphate solvent at temperatures exceeding 135°C.[9] Under optimum conditions, the reaction becomes explosive,thus the evaporator temperature must be limited to avoid explosive conditions.

Source Term

The evaporator would typically contain about 2000 gallons of uranyl nitrate.[7] At a uranyl nitrate density of 2.8 gms/cm^3, the total mass of uranium in the evaporator would be about 10,000 kg. Adjacent to the evaporator, an equal amount of product might be stored in a surge tank.

The consequences of explosion accidents are limited by the material that can be maintained in the air rather than by the total volume or mass of material involved in the explosion. The airborne concentration of heavy particles in the respirable range appears to be limited to approximately 100 mg/m^3.[8] The material splattered on the walls and floors which subsequently becomes airborne as it dries is expected to constitute an insignificant fraction of the original source term. Thus for a room volume assumed to be approximately $10^4 m^3$, the quantity of uranium released to the environment is estimated to be approximately 1000 kg.

It is assumed that the milling operation has removed most of the uranium daughter products in the ore, so that nearly all of the activity released in the evaporator explosion is from the isotopes of natural uranium. Their specific activities are 3.31×10^{-7} Ci/gm of ^{238}U, 1.54×10^{-8} Ci/gm of ^{235}U, 3.52×10^{-7} Ci/gm of ^{234}U, and 3.31×10^{-7} Ci/gm of ^{234}Th.

Likelihood

Historically, one explosion associated with evaporator operation has occurred due to a "red-oil" reaction,[7] but this was not in a commercial uranium hexafluoride conversion facility.

The probability of a red-oil explosion in the low activity waste concentration of a commercial fuels reprocessing plant has been estimated to be $\sim 10^{-4}$/plant-year.[9] The likelihood of a chemical explosion in a fuel fabrication plant has been estimated to be $\sim 10^{-3}$ plant-year.[8]

Estimate: Source term $\approx 3.3 \times 10^{-1}$ *Ci* ^{238}U, 1.5×10^{-2} *Ci* ^{235}U, 3.5×10^{-1} *Ci* ^{234}U, 3.3×10^{-1} *Ci* ^{234}Th *to air. Likelihood* $\approx 10^{-3}$ *to* 10^{-4}/*plant-year.*

A.2 Hydrogen Explosion in the Reduction Step

In either the wet chemical solvent extraction or the hydrofluor process for uranium hexafluoride conversion, the uranium concentrate is reduced to UO_2 utilizing dissociated ammonia ($N_2 + H_2$) as the reductant. Although the hydrogen evolved from the cracked ammonia is diluted to below the explosive concentration, the potential for an explosion must always be considered when hydrogen is used in the process.

Source Term

It is assumed that the reductor contains a uranium inventory of approximately 10,000 kg. As discussed in the case of the uranyl nitrate evaporator explosion, the consequences of explosion accidents are limited by the concentration of heavy material that can be maintained in the air, shown to be approximately 100 mg/m^3.[8] Then for a room volume assumed to be of the order of $10^4 m^3$, the quantity of uranium released to the environment is estimated to be approximately 1000 kg. The isotopic composition of the material released is taken to be that of freshly separated natural uranium.

Likelihood

No incidents have been recorded involving hydrogen explosions in the reduction step of uranium hexafluoride conversion.[16] Hydrogen fires or explosions have been documented, however, in association with fuel processing activities.[31,32] An estimate of the probability of a hydrogen explosion in the sintering step of fuel fabrication is $\lesssim 5 \times 10^{-2}$/plant-yr.[8] The likelihood of a chemical explosion in a fuel fabrication plant has been estimated to be $\sim 10^{-3}$/plant-yr.[8]

Estimate: Source term $\approx 3.3 \times 10^{-1}$ *Ci* ^{238}U, 1.5×10^{-2} *Ci* ^{235}U, 3.5×10^{-1} *Ci* ^{234}U, 3.3×10^{-1} *Ci* ^{234}Th *to air. Likelihood* $\approx 5 \times 10^{-2}$ *to* 10^{-3}*/plant-yr.*

B.1 Fire in the Solvent Extraction Operation

In the wet chemical solvent extraction process for producing uranium hexafluoride, the uranium concentrate is dissolved in nitric acid and sent to the uranium-extraction column, where the aqueous solution of uranyl nitrate is extracted countercurrently with tributyl phosphate in hexane. The solvent extraction operation is likely to be carried out in a separate building consisting of two operating parts. The first is a solvent rework section where most of the organic solvent used in the process, hexane, is stored and prepared for use. The other is the solvent extraction section where the uranium purification operation is carried out. The flammability of the solvent provides the potential for a serious fire in either section, however, only a fire in the solvent extraction circuit would involve the release of radioactivity.

Source Term

It is estimated that the amount of loaded solvent which might typically be involved in a fire in the solvent extraction section is approximately 2500 gallons containing 800 kg of uranium.[7]

If 1% of the uranium were dispersed, as assumed in estimates of the amount of plutonium dispersed in a solvent extraction fire,[9] 8 kg of uranium could be released to the atmosphere. If 0.5-0.7% were dispersed, as assumed in estimates of the amount of plutonium released from a fire near nitrate blending tanks,[8*] 4 to 6 kilograms of uranium could be released to the atmosphere.

It is assumed that the thorium and 1% of the radium in equilibrium with natural uranium is present in this initial purification step. Thus the estimated concentrations of radioisotopes in the material released are: 3.31×10^{-7} Ci/gm of ^{238}U, 1.54×10^{-8} Ci/gm of ^{235}U, 3.52×10^{-7} Ci/gm of ^{234}U, 3.31×10^{-7} Ci/gm of ^{234}Th, 3.31×10^{-7} Ci/gm of ^{230}Th, and 3.31×10^{-9} Ci/gm of ^{226}Ra.

Likelihood

There have been no solvent extraction fires on record associated with uranium hexafluoride production.[31,32] However, from chemical industry data, the probability of major fires per plant-year is estimated to be 4×10^{-4}.

Estimate: Source term $\approx 2.0 \times 10^{-3}$ *Ci* ^{238}U, 9.2×10^{-5} *Ci* ^{235}U, 2.1×10^{-3} *Ci* ^{234}U, 2.0×10^{-3} *Ci* ^{234}Th, 2.0×10^{-3} *Ci* ^{230}Th, 2.0×10^{-5} *Ci* ^{226}Ra, *to air; Likelihood* $\approx 4 \times 10^{-4}$*/plant-yr.*

C.1 Release from a Hot UF_6 Cylinder

At the tail end of the uranium hexafluoride production process, the liquid uranium hexafluoride is transferred under pressure to a large cylinder,

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[*] Experiments have demonstrated that approximately 80% of the uranium aerosol generated from a nitrate solution involved in a gasoline fire is in the respirable size range.[8]

nominally either 14 or 10 tons. The cylinder is placed in a steam-heated chest for about 12 hours at 200°F to homogenize the contents. A valve is installed and a sample taken. The cylinder is then removed by fork lift to an outdoor storage area where the UF_6 cools and solidifies in approximately 72 hours. At any time during this sequence of events, while the UF_6 is in the liquid state, the failure of a valve, an operator error, or a cylinder rupture could release significant quantities of UF_6 to the environment.

Source Term

If a large cylinder containing 12.7 MT of UF_6 were to rupture or lose a valve, and if the release were to go unchecked until the contents of the cylinder solidified, analyses result in estimates of approximately 2800 kg of uranium released to the environment.[7] In fact, the rupture of a ten ton cylinder at an enrichment plant, resulting in the release of 3077 kg of uranium, is a matter of record.[31] However, the most probable release in the event of a breach in containment is not likely to coincide with the maximum release. A summary of releases associated with hookup and disconnect operations on UF_6 cylinders at the three existing gaseous diffusion plants has been compiled and is reproduced in Table 4-12.[38,39] The range of release falls between approximately 10 and 400 kg uranium in the period 1969 through 1973. The average release in six recorded incidents resulting in more than 5 kg uranium loss was 108 kg of uranium. The uranium at the tail end of the uranium hexafluoride conversion process is freshly separated from daughter products.

TABLE 4-12

UF_6 RELEASES (> 5Kg) ASSOCIATED WITH FILLING OR FEEDING CYLINDERS AT THE GASEOUS DIFFUSION PLANTS IN THE PERIOD 1969-1973

Year	Plant	Quantity (KgU)	Reason
1969	Portsmouth	400	Tails cylinder valve would not close
1969	Portsmouth	9.9	Leaking pigtail
1970	Oak Ridge	153.3	Pigtail connection leak in normal assay feed autoclave
1970	Portsmouth	12	Feed position pigtail gasket failure
1971	Paducah	14.7	Pigtail rupture during sampling of cylinder
1973	Portsmouth	59.4	Undetected plugged and open valve on hot cylinder

Likelihood

There are no UF_6 cylinder releases on record in the ~ 20 plant-years of operation of commercial uranium hexafluoride conversion plants.[40,41]

From failure data on valve ruptures,[6] the likelihood of a UF_6 cylinder valve rupture while the UF_6 is in the liquid state is estimated to lie in the range, 5×10^{-5} to 5×10^{-3}/plant-yr for a 5000 MTU/yr plant capacity. However, none of the releases associated with the filling or feeding of cylinders at an enrichment plant, as shown in Table 4-12, can be attributed to valve ruptures.

From the incidents on record at the three enrichment facilities, a rough estimate of the likelihood of UF_6 cylinder releases can be made which may be extrapolated to filling operations at a UF_6 conversion facility. In 1973, roughly 7800 UF_6 cylinder hookup and disconnect operations were performed at the Paducah plant, 6700 at the Oak Ridge plant, and 10,000 at the Portsmouth plant. Then for the six releases on record documented in Table 4-12, and assuming the same level of operations over the entire 5-year period, the likelihood of a release in handling a cylinder is approximately 5×10^{-5}/cylinder.

The generic UF_6 conversion plant operating at a nominal capacity of 5000 MTU/yr would require the filling of about 580 large product cylinders annually. Using the release probabilities developed in the previous discussion from the enrichment plant data, the likelihood of a release is approximately 3×10^{-2}/plant-year.

Estimate: Source term $\approx 3.6 \times 10^{-2}$ Ci ^{238}U, 1.7×10^{-3} Ci ^{235}U, 3.9×10^{-2} Ci ^{234}U, 3.6×10^{-2} Ci ^{234}Th to air. Likelihood $\approx 3 \times 10^{-2}$/plant-year.

C.2 Valve Failure in the Distillation Step

In the dry hydrofluor process for the conversion of uranium hexafluoride, the final step of process involves fractional distillation. The distillation step removes volatile fluorides generated in the fluorination step. The failure of a valve or piping in this step could result in the release of UF_6 to the environment.

Source Term

In an incident of record, a valve bonnet failed in a line from a vaporizer tank in the distillation area of a plant and released approximately 40 kg of uranium as UF_6 to the building.[7]

Small releases of UF_6 also occur in the enrichment plants from time to time due to defective valves or tubing, and the parallel to the distillation step of a conversion plant can be drawn. Thus it is interesting to note that in 13 such incidents recorded at the diffusion plants over the past 20 years, an average of about 44 kg of UF_6, or approximately 30 kg of uranium, was released.[42]

Likelihood

At least one accident on record involved the failure of a valve bonnet in the distillation area of a UF_6 plant.[7] Using the 20 plant-years of commercial conversion plant operation as a base, the one incident results in a likelihood of ~ 5 x 10^{-2}/plant-year.

Another approach to estimating the likelihood of such an accident is to use the failure data on valve ruptures, 10^{-9} to 10^{-7}/hr.[6] Assuming 300 day around-the-clock operation, and the order of 10 to 100 valves in the process, one arrives at a range of 10^{-4} to 10^{-1} valve failures/plant yr. Pipe failure likelihoods are an order of magnitude lower on a per section basis,[6] and thus can be neglected in comparison with valve failures.

__Estimate__: 1.3 x 10^{-2} *Ci* ^{238}U, *6.2 x* 10^{-4} *Ci* ^{235}U; , *1.4 x* 10^{-2} *Ci* ^{234}U, *1.3 x* 10^{-2}*Ci* ^{234}Th *to air; Likelihood* $\approx$ *5 x* 10^{-2}*/plant-yr.*

E.1 Release of Raffinate from the Waste Retention Pond

In the wet chemical solvent extraction process for producing uranium hexafluoride, the raffinate stream from the solvent extraction step must be permanently disposed of. Currently, the raffinate stream is combined with miscellaneous liquid streams from the process, such as spent scrubber solutions, is neutralized by ammonia, and permanently impounded in earthen-walled retention basins. The impounded liquid is composed of ammonium nitrate, nitric acids, and metallic salts, as well as soluble radionuclides. If a dike failure were to occur, as much as eight million gallons of contaminated water could be released to the watercourse.[7]

Source Term

The measured concentrations of radionuclides in the waste retention pond are given in Table 4-13.[7]

There have been no recorded major releases in the four to five years of operation of the UF_6 conversion waste retention pond, although minor seepage has been noted.[41] Accordingly, the release data related to uranium mill tailings ponds will be cited (see Section 4.2, Accident E.1). From the tailings pond data, the average releases to the watercourse from dike failure or flooding are 3000 m^3 of liquids and 7.5×10^6 lbs of solids. The solids, however, are assumed to deposit in the vicinity of the entry point to the watercourse.

Likelihood

There has been no major release recorded during the four to five years of operation of the UF_6 conversion waste retention pond.[41] From the mill tailings pond data (see Section 4.2, accident E.1), the likelihood of a release involving the source term described in the previous section is approximately 2×10^{-2}/plant-yr of mill operation. As pointed out in Section 4.2, however, this estimate may be high, since the evaluation of early dike construction led to strenghtening of existing dikes and improved construction standards.

Estimate: Source term $\simeq 1.7 \times 10^{-4}$ *Ci* ^{238}U, 8.0×10^{-6} *Ci* ^{235}U, 1.8×10^{-4} *Ci* ^{234}U, 1.7×10^{-4} *Ci* ^{234}Th, 3.0×10^{-3} *Ci* ^{226}Ra, 3.0×10^{-5} *Ci* ^{230}Th, *to watercourse. Likelihood* $\lesssim 2 \times 10^{-2}$*/plant-yr.*

TABLE 4-13

RADIONUCLIDE CONCENTRATIONS IN UF_6 CONVERSION WASTE RETENTION POND[7]

Radionuclide	Concentration (μCi/ml)
Ra-226	1×10^{-6}
Th-230	1×10^{-8}
Uranium	1.2×10^{-7}

4.4 Enrichment

The accidents considered in enrichment facilities, keyed to the accident categories given in Table 4-1, are listed in Table 4-14.

Small releases resulting from occasional leaks in piping or valves are considered to be credited to the normal operation of the plant. The potentially toxic effects resulting from the postulated release of anhydrous hydrogen fluoride is considered outside of the scope of the current study. Natural or man-induced disasters, such as tornados, earthquakes, floods, or missile impacts are highly unlikely, although their occurrence could result in varying releases of activity. The risks associated with these events were not analyzed in the current study.

Table 4-15 provides historical data on the three gaseous diffusion enrichment complexes currently in operation.[43]

B.1 Catastrophic Fire

Gaseous UF_6 from leaks in the process stages can combine with oil vapors from pumps or other machinery if the ventilation system allows these vapors to accumulate in parts of the process building. The reaction is explosive, following an equation of the type:

$$[CH_2]_3 + UF_6 + 4O_2 \longrightarrow 6HF + UO_2 + 3CO_2,$$

or others in which UO_2F_2 and H_2O can be generated. The heat generated can melt the roof and floor structures, and release the uranium in process in the stages involved in the fire.

TABLE 4-14

ENRICHMENT PLANT ACCIDENTS

B.1	Catastrophic fire
C.1	Release from a hot UF_6 cylinder
C.2	Significant leaks or failure of valves or piping within the plant
D.1	Criticality

TABLE 4-15

EXISTING ENRICHMENT PLANT DATA[43]

Plant	Capacity (millions of kg SWU in 1970)	Enrichment Range (% U-235 in 1970)	Completion Dates	Years of Operation
Oak Ridge	4.73	0.7-4	1945-1954	~31
Paducah	7.31	0.7-1.1	1953-1954	~23
Portsmouth	5.19	0.7-97.7	1955-1956	~21
			Total	~75

Source Term

Although catastrophic fires have occurred in enrichment plants, resulting in significant property damage,[31] estimates of the resulting quantity of uranium released to the environment are not available.

At least one analytical estimate has been made of the consequences of a catastrophic fire in an enrichment plant.[16] It was assumed that the contents of the 16 stages comprising a cell are released to the environment. The resulting release of uranium was estimated to be 1550 kg, with a composition characteristic of the feed.

In determining the isotopic composition of the feed, it was assumed that in the generic enrichment plant, 90% of the feed is natural uranium and 10% recycled uranium from reprocessing LWR fuel. Traces of fission products are also carried along with the recycled uranium, but their activity is sufficiently low to be neglected. Thus the estimated isotopic activities of the feed are given in Table 4-16.

Likelihood

From chemical industry data, the probability of major fires is estimated to be 4 x 10^{-4}/plant-year.[8] However, at least two major fires, each resulting in property damage in excess of $2 million, have occurred at the Paducah enrichment plant.[31] Referring to Table 4-15, there have been ~75 plant-years of enrichment plant operating experience. Thus, from the historical incidents, the likelihood of a major fire in the enrichment plant is approximately 3 x 10^{-2}/plant-year.

TABLE 4-16

ISOTOPIC ACTIVITIES OF THE ENRICHMENT PLANT FEED

Isotope	Activity per Gram of Uranium (Ci)
^{238}U	3.31×10^{-7}
^{237}U	2.43×10^{-7}
^{236}U	3.02×10^{-8}
^{235}U	1.57×10^{-8}
^{234}U	4.06×10^{-7}
^{234}Th	3.31×10^{-7}

Estimate: *Source term* $\approx$ *5.1 x* 10^{-1} *Ci* ^{238}U, *3.8 x* 10^{-1} *Ci* ^{237}U, *4.7 x* 10^{-2} *Ci* ^{236}U, *2.4 x* 10^{-2} *Ci* ^{235}U, *6.3 x* 10^{-1} *Ci* ^{234}U, *5.1 x* 10^{-1} *Ci* ^{234}Th *to air; Likelihood* $\approx$ *3 x* 10^{-2} *to 4 x* 10^{-4}*/plant-year.*

C.1 Release from a Hot UF_6 Cylinder

Natural uranium hexafluoride from the conversion facility and recycle uranium hexafluoride from the reprocessing plant arrive at the enrichment facility in cylinders which must be heated in autoclaves in order to effect the transfer of material. This takes place at the feed vaporization operation. Valve rupture at this stage would have no other consequences than a negligible release inside the building because the UF_6 is in the solid phase. Once the cylinder has been heated in the sealed autoclave, significant releases of UF_6 outside of the process containment system are not possible. Thus accidents involving feed cylinders are not credible sources of significant environmental releases. In contrast, product cylinders, containing 2.2 MT of UF_6, and tails cylinders, containing up to 12.7 MT of UF_6, are carried out of the autoclave to a cylinder storage area while they are still hot. The failure of a valve, an operator error, or rupture of the cylinder itself while the contents are still in the liquid phase could release significant quantities of UF_6 to the environment.

Source Term

A summary of the releases over 5 kg associated with the filling or feeding of UF_6 cylinders at the diffusion plants over the period 1969-1973 was presented in Table 4-12.[38,39] The range of releases falls between approximately 10 and 400 kg uranium. The average release in six recorded incidents was 108 kg of uranium.

The predicted specific activities by isotope of product and waste cylinders, assuming that 90% of the feed is natural uranium and 10% is recycled uranium, and that the product ^{235}U enrichment is 2.6% and the tails composition is 0.25% ^{235}U, are given in Table 4-17. Although the type of cylinder involved in the release is not specified in the historical release data given in Table 4-12, it is estimated that an enrichment facility operating at a capacity of 8.75×10^6 kg SWU/yr would require the filling of approximately 2000 product cylinders and 1300 waste cylinders. Assuming that the likelihood of a release from a particular cylinder is proportional to the number of operations involving that type of cylinder, the weighted averages of the specific activities given in Table 4-17 are as follows: 3.28×10^{-7} Ci/gm of ^{238}U, 2.92×10^{-7} Ci/gm of ^{237}U, 4.86×10^{-8} Ci/gm of ^{236}U, 3.55×10^{-8} Ci/gm of ^{235}U, and 1.40×10^{-6} Ci/gm of ^{234}U.

Likelihood

From the historical data on releases at the gaseous diffusion facilities summarized in Table 4-12, a release probability of 0.4/plant-year is derived for the six releases on record.

Estimate: Source term $\approx 3.5 \times 10^{-2}$ *Ci* ^{238}U, 3.2×10^{-2} *Ci* ^{237}U, 5.3×10^{-3} *Ci* ^{236}U, 3.8×10^{-3} *Ci* ^{235}U, 1.5×10^{-1} *Ci* ^{234}U *to air; Likelihood* $\approx 4 \times 10^{-1}$*/plant-year.*

TABLE 4-17

SPECIFIC ACTIVITIES BY ISOTOPE OF PRODUCT AND TAILS CYLINDERS AT THE ENRICHMENT PLANT

Isotope	Activity (Ci/gram Uranium)	
	Product (2.2 MT Cylinder)	Tails (12.7 MT Cylinder)
^{238}U	3.24×10^{-7}	3.33×10^{-7}
^{237}U	3.73×10^{-7}	1.71×10^{-7}
^{236}U	7.11×10^{-8}	1.48×10^{-8}
^{235}U	5.56×10^{-8}	5.39×10^{-9}
^{234}U	2.26×10^{-6}	9.67×10^{-8}

C.2 Significant Leaks or Failure of Valves or Piping Within the Plant

Approximately 1100 stages are required for the enrichment of natural uranium to 2.6% ^{235}U for a tails assay of 0.25%. Each stage requires extensive piping and valves to route the compressed gas through the cascade. Releases of UF_6 occur from time to time due to valve or piping failures, or significant leaks.

Source Term

An average of about 2.4 Kg of uranium was released in the plant from significant leakage or the failure of valves or piping in the three gaseous diffusion facilities over the period 1969 through 1973.[38,39] It is assumed that the isotopic composition of uranium released in such incidents was characteristic of the feed (see Table 4-16).

Likelihood

There were 27 recorded incidents of releases from the three gaseous diffusion facilities in the period 1969 through 1973 resulting in the release of greater than 100 grams of uranium. On this basis, assuming that the release probability is independent of the plant capacity, the likelihood of such releases is approximately 1.8/plant-year.

Estimate: Source term $\approx 7.9 \times 10^{-4}$ *Ci* ^{238}U, 5.8×10^{-4} *Ci* ^{237}U, 7.3×10^{-5} *Ci* ^{236}U, 3.8×10^{-5} *Ci* ^{235}U, 9.7×10^{-4} *Ci* ^{234}U, 7.9×10^{-4} *Ci* ^{234}Th *to air; Likelihood* $\approx$ *1.8/plant-year.*

D.1 Criticality

Although process configurations and administrative procedures are carefully designed and monitored with criticality control in mind, the possibility of the formation of a critical mass, accompanied by the evolution of ionizing radiation and fission products, is always a possibility when handling enriched uranium. For the low enrichments involved in LWR fuel processing, and the low densities of uranium handled in the enrichment operation, inadvertant formation of a critical mass is, indeed, a remote possibility. Water moderation would be required accompanied by a rearrangement in configuration, both of which are prevented by deliberate design and administrative procedures.

Source Term

For the 26 criticality incidents on record,[31] the total number of fissions range from $\sim 3 \times 10^{15}$ to 4×10^{19}. For the 11 non-solution criticalities, considered more representative of a postulated criticality incident for enrichment, the average number of fissions is $\sim 10^{17}$. The radionuclides released ten minutes after an incident involving 10^{17} fissions, based upon ORIGEN calculations,[44] are given in Table 4-18. These estimates further assume 100% of the noble gases, 50% of the halogens, and 0.2% of the actinides released to the environment. Additionally, the neutron and gamma radiation associated with a burst of 10^{17} fissions would result in a dose to the population. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

TABLE 4-18

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT AT THE ENRICHMENT PLANT

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	1.2×10^{-2}	I 136	2.3
Br 80m	5.7×10^{-6}	I 137	3.3×10^{-5}
Br 82	1.9×10^{-6}	Xe 133	1.7×10^{-4}
Br 82m	3.2×10^{-4}	Xe 133m	9.9×10^{-6}
Br 83	3.5×10^{-1}	Xe 135	2.2×10^{-1}
Br 84	3.5	Xe 135m	4.7×10^{-1}
Br 84m	1.5×10^{-1}	Xe 137	9.0×10^{1}
Br 85	7.2	Xe 138	7.0×10^{1}
Br 86	1.5×10^{-1}	Xe 139	1.5×10^{-1}
Br 87	2.1×10^{-1}	Xe 140	2.2×10^{-8}
Kr 83m	3.4×10^{-2}	Th 231	2.0×10^{-9}
Kr 85	3.2×10^{-6}	Th 234	2.2×10^{-10}
Kr 85m	1.3	Pa 234m	1.8×10^{-10}
Kr 87	9.4	U 233	1.7×10^{-16}
Kr 88	6.3	U 234	3.4×10^{-14}
Kr 89	5.2×10^{1}	U 235	4.5×10^{-7}
Kr 90	9.2×10^{-3}	U 236	1.3×10^{-12}
I 128	1.4×10^{-4}	U 237	7.1×10^{-6}
I 130	1.0×10^{-4}	U 238	1.1×10^{-6}
I 131	5.5×10^{-3}	U 239	6.1×10^{-1}
I 132	1.6×10^{-1}	Np 237	4.4×10^{-17}
I 133	1.5×10^{-1}	Np 239	1.5×10^{-3}
I 134	5.2	Np 240	3.5×10^{-12}
I 135	2.3	Pu 239	4.1×10^{-13}

Likelihood

There have been no criticality incidents in the 75 plant-years of uranium enrichment. Moreover, there has not been a criticality incident recorded involving low enrichment uranium. The only criticality incidents on record involved the handling of plutonium, highly enriched uranium ($\gtrsim$ 83% enriched in ^{235}U), or ^{233}U.[31]

An estimate has been made of the probability of criticality in fuel fabrication plants.[8] From the four recorded incidents, and the estimated 432 (increased to ~ 490 through 1975) plant-years of production involving uranium and plutonium fuel fabrication, a probability of $\sim 8 \times 10^{-3}$ criticality accidents/plant-year has been derived. As discussed in Reference 8, an improved basis for such an estimate would consider the total fuel throughput, the fuel forms during processing, and the fuel reactivities involved.

The only criticalities on record in fuel fabrication or reprocessing facilities occurred in solution. Dry criticalities have only occurred in reactor experiments. Since enrichment operations are dry, the likelihood of criticality is estimated to be at least an order of magnitude lower than the above estimate. Moreover, the low enrichment associated with LWR fuel is estimated to reduce the likelihood of criticality by at least another order of magnitude.

Estimate: *Source Term - Radionuclides given in Table 4-18 to air plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood* $\lesssim 8 \times 10^{-5}$*/plant-year.*

4.5 Uranium Fuel Fabrication

The accidents considered in uranium fuel fabrication facilities, keyed to the accident categories given in Table 4-1, are listed in Table 4-19.

The risk associated with other incidents, some of which have occurred in the past, is judged to be insignificant in comparison with the accidents considered in Table 4-19. These include spills of UO_2 powder, sintering furnace explosions, autoclave explosions, inadvertent release of liquid wastes, and ventilation problems from, for example, loss of electrical power. These have resulted in excessive airborne concentrations in work areas, but the release to the environment is generally inconsequential. Filter failures in the process ventilation stream have occurred from time to time,[32] but the filters are generally replaced before the time integrated release becomes a significant fraction of the annual release from normal operations.

A tornado which is capable of demolishing the building structures could disperse significantly large quantities of respirable uranium to the environment,but the risk associated with such an accident has not been evaluated due to lack of pertinent data.

A list of government contractors and commercial firms engaged in fuel fabrication activities since 1942 is given in Table 4-20.[8] The levels of production and the nature of the processes vary substantially. Also, some of the commercial firms have conducted activities at more than one plant, although they are listed only once in the Table. Nevertheless,

TABLE 4-19

URANIUM FUEL FABRICATION ACCIDENTS

A.1	Hydrogen explosion in reduction furnace
B.1	Major facility fire
B.2	Fire in a roughing filter
C.1	Release from a hot UF_6 cylinder
C.2	Failure of valves or piping within the plant
D.1	Criticality
E.1	Waste retention pond failure

TABLE 4-20

ESTIMATE OF PLANT-YEARS OF PRODUCTION SINCE 1942 INVOLVING FUEL FABRICATION[8]

Plant	Estimated Dates	Estimated Plant-Years
Hanford	1944-1975	31
Savannah River Laboratory	1954-1975	21
Los Alamos Scientific Laboratory	1943-1975	32
National Lead Company of Ohio	1944-1975	31
Oak Ridge National Laboratory	1943-1975	32
Lawrence Radiation Laboratory	1949-1975	26
Argonne National Laboratory	1949-1975	26
Aerojet General Nuclear	1955-1970	15
Atomics International	1955-1975	20
Babcock & Wilcox	1957-1975	18
Clevite Research Corporation	1957-1969	12
Combustion Engineering	1955-1975	20
Curtiss-Wright Davison	1955-1975	20
Gulf General Atomic	1958-1975	17
General Electric	1955-1975	20
Gulf United Nuclear	1971-1975	4
M & C Nuclear, Incorporated	1961-1965	4
Exxon	1971-1975	4
Mallinckrodt Nuclear Corporation	1960-1970	10
Martin Company	1960-1970	10
Kerr-McGee	1969-1975	6
National Carbon Company	1960-1965	5
National Lead Company	1962-1975	13
Engelhard Industries, Incorporated	1957-1970	13
Nuclear Development Corporation of America	1957-1968	11
Nuclear Materials and Equipment Corporation	1960-1971	11
Sylvania-Corning Nuclear Corporation	1960-1968	8
Westinghouse Electric Corporation	1955-1975	20
United Nuclear	1957-1975	18
U.S. Nuclear	1972-1975	3
Nuclear Fuel Services	1966-1975	9
	Total	490

the total number of plant-years may be used for rough estimates of accident probabilities from historical records on incidents.

A.1 Hydrogen Explosion in Reduction Furnace

In the process for producing uranium dioxide from uranium hexafluoride, the gaseous UF_6 is hydrolyzed to uranyl fluoride and reacted with ammonia to precipitate ammonium diuranate (ADU). The ADU slurries are concentrated and then calcined to form U_3O_8. The U_3O_8 is reduced to uranium dioxide at a temperature of approximately 1000°F in a reducing atmosphere of hydrogen.* The hydrogen concentration is controlled to prevent the buildup of an explosive atmosphere. However, should these controls fail, the hydrogen could ignite and explosively blow the uranium out of the furnace.

Source Term

The rotary kiln reduction furnace would typically contain in excess of 100 kg uranium dioxide. The consequences of explosion accidents are limited by the material that can be maintained in the air rather than by the total volume or mass of material involved in the explosion. The airborne concentration of heavy particles in the respirable range appears to be limited to approximately 100 mg/m^3.[8] Thus for a room volume assumed to be approximately $10^4 m^3$, the quantity of uranium released to the building ventilation system would be approximately 1 kg of uranium.

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* A reducing atmosphere is also used later in the fabrication process during the sintering of pellets. However, by virtue of the integral form of the uranium dioxide at this stage of the process, the consequences of a postulated explosion in the sintering furnace is judged to be insignificant in comparison with a similar event in the reduction furnace.

Uranium fuel fabrication plants have at least one high efficiency particulate air (HEPA) filter in the process ventilation system. However, the existence of HEPA filters in the building air ventilation system is not assured. Some of the existing plants have one HEPA filter in the building ventilation system and others have none.* It is not clear at the present time whether new uranium fuel fabrication plants will be required to incorporate a HEPA filter in the building ventilation system. The efficiency of a single HEPA filter against particles in the respirable size range is taken to be 99.9%.[8] However, because of the variability in the design of building ventilation systems, a source term range is adopted for this accident of 1 to 1000 gms uranium. The isotopic composition of the uranium is taken to be that of the product from the enrichment plant, given in Table 4-17.

Likelihood

An estimate of $\lesssim 5 \times 10^{-2}$/plant-year has been made for the likelihood of a hydrogen explosion in a fuel fabrication sintering furnace.[8]** The same study derived an estimate of $\sim 10^{-3}$/plant-year for chemical explosions in general.

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* Only one plant has two HEPA filters in the building air ventilation system.

** The facility associated with this estimate was a 300 MT HM/yr mixed oxide plant which possesses a significantly lower annual furnace throughput than the generic uranium fuel fabrication plant selected for this study.

At least one accident has occurred (at a sintering furnace) resulting from detonation of an explosive mixture of hydrogen and oxygen.[8,31] On the basis of the ~490 plant-years of fuel fabrication experience, this would result in a likelihood of $\gtrsim 2 \times 10^{-3}$/plant-year.

In another accident evaluation,[45] the probability of a hydrogen explosion in the sintering furnace has been crudely estimated to be lower than 10^{0} to 10^{-1}/year and higher than 10^{-1} to 10^{-3}/year.

__Estimate:__ Source term $\approx 3.2 \times 10^{-4}$ *to* 3.2×10^{-7} *Ci* ^{238}U, 3.7×10^{-4} *to* 3.7×10^{-7} *Ci* ^{237}U, 7.1×10^{-5} *to* 7.1×10^{-8} *Ci* ^{236}U, 5.6×10^{-5} *to* 5.6×10^{-8} *Ci* ^{235}U, 2.3×10^{-3} *to* 2.3×10^{-6} *Ci* ^{234}U *to air; Likelihood* $\approx 5 \times 10^{-2}$ *to* 2×10^{-3}*/plant-year.*

B.1 Major Facility Fire

Since combustibles are limited in a fuel fabrication facility, the occurrence of a major facility fire is remote, but could conceivably occur. It could be initiated, for example, by the ignition of a solvent container which is improperly handled. The fire could then spread to a large fraction of the process enclosures in several parallel fabrication lines.

Source Term

The total amount of UO_2 in-process in the powder and pellet treatment steps of fabrication, when the material is in the dispersible form, may be on the order of 5000 kg. However, as much as 200 MT of UO_2 powder may be stored in racks at the facility, and approximately 50% of this powder could be in the dispersible size range.

It is estimated that, in the event of a major facility fire, as much as 1% of the dispersible UO_2 powder could be rendered airborne.[8,42] The exhaust fans would continue to operate, sweeping the powder out through the ventilation system. However, the final HEPA filter barrier, if available, would continue to function with an estimated efficiency of 99.9%.[8] As discussed in the previous section, the variability in building ventilation system designs requires the consideration of a range for the source term. Thus the amount of uranium released to the environment from this accident is estimated to be in the range of 1 to 1000 kg. The isotopic composition of the uranium is taken to be that of the product from the enrichment plant, given in Table 4-17.

Likelihood

The likelihood of a major fire in a mixed oxide fabrication plant, which is similar in design to the uranium fabrication facility considered here, is estimated to be $\sim 2 \times 10^{-4}$/plant-year. Although a catastrophic facility fire has occurred at the Rocky Flats facility in 1969,[31] this incident is not considered germane to the current estimate, since metallic plutonium was the probable source of ignition.[42]

Estimate: *Source term* $\approx 3.2 \times 10^{-1}$ *to* 3.2×10^{-4} *Ci* ^{238}U, 3.7×10^{-1} *to* 3.7×10^{-4} *Ci* ^{237}U, 7.1×10^{-2} *to* 7.1×10^{-5} *Ci* ^{236}U, 5.6×10^{-2} *to* 5.6×10^{-5} *Ci* ^{235}U, *2.3 to* 2.3×10^{-3} *Ci* ^{234}U *to air; Likelihood* $\approx 2 \times 10^{-4}$*/plant-yr.*

B.2 Fire in a Roughing Filter

Roughing filters are installed in the exhaust plenums from dusty operations, such as the oxide milling station, in the fuel fabrication plant.

In addition to the uranium oxide powder, small amounts of lint, lubricating materials, or other combustible materials may be trapped in these filters. A fire could be started by an electrical spark or static electricity, resulting in the destruction of the filter and the release of its entire uranium inventory.

Source Term

It is assumed that the inventory in the enclosure surrounding the ball mill served by the roughing filter is approximately 25 kilograms of uranium dioxide. Although the maximum filter loading may be as high as 5% of the total enclosure inventory,[16] it is assumed that the nominal filter loading is 1% of the inventory, all in the respirable range. The fire is assumed to release to the building air all of the powder trapped within the filter.

As discussed earlier, if a HEPA filter is included within the building ventilation system, a reduction in the amount of uranium released to the environment by 99.9% is estimated.[8] However, since many fabrication plants do not incorporate HEPA filters in the building air exhaust systems, a range of values is selected for the source term. Thus the quantity of uranium released to the environment from this accident is estimated to fall within the range of 0.25 to 250 gms. The isotopic composition of the uranium is taken to be that of the product from the enrichment plant, given in Table 4-17.

Likelihood

The likelihood of a local fire in a mixed oxide fuel fabrication facility has been estimated to be $\lesssim 10^{-2}$/plant-year. According to the incidents on record, at least five local fires (exclusive of ion-exchange resin fires) have occurred in the 490 plant-years of fuel fabrication activities.[32]

Estimate: *Source term* $\approx 8.1 \times 10^{-5}$ *to* 8.1×10^{-8} *Ci* ^{238}U, 9.3×10^{-5} *to* 9.3×10^{-8} *Ci* ^{237}U, 1.8×10^{-5} *to* 1.8×10^{-8} *Ci* ^{236}U, 1.4×10^{-5} *to* 1.4×10^{-8} *Ci* ^{235}U, 5.7×10^{-4} *to* 5.7×10^{-7} *Ci* ^{234}U *to air; Likelihood* $\approx 10^{-2}$*/plant-year.*

C.1 Release from a Hot UF_6 Cylinder

Enriched uranium is received at the fabrication facility in a cylinder containing 2.2 MT of uranium hexafluoride. The cylinder is placed in a steam-heated chest, where the uranium hexafluoride is vaporized and passed directly to the first step of the conversion process. Should the cylinder be overpressurized, a rupture could cause the entire contents of the cylinder to be released into the vaporization room. A more probable accident would be the development of a leak, or an operator error resulting in an inadvertently backed out valve, in which case a portion of the liquid would evaporate to replace the escaping gas. Evaporation would continue until enough heat is removed by the escaping gas, or by forced cooling through a water-spray protective system, to cause the remaining UF_6 to solidify.

Source Term

If a cylinder containing 2.2 MT of UF_6 were to rupture or lose a valve, and if the release were to go unchecked until the contents of the cylinder solidified, analyses result in estimates of approximately 700 kg of uranium released to the vaporization room.[7] However, the most probable release is not likely to coincide with the maximum release. A summary of releases associated with hookup and disconnect operations at the gaseous diffusion plants in the period 1969 through 1973 is given in Table 4-12. The average release in six recorded incidents resulting in more than 5 kg uranium loss was 108 kg of uranium. Since the data base associated with enrichment is more extensive than that associated with fabrication operations, and since the UF_6 cylinder hookup and disconnect operations are similar, the average source term from enrichment will be adapted here.

It is further assumed that the HEPA filter in the process ventilation system becomes plugged with hydrolized UF_6, allowing the UF_6 and its reaction products, UO_2F_2 and HF, to seep from the vaporization room. If a HEPA filter is incorporated within the building ventilation system, a reduction in the amount of uranium released to the environment by 99.9% is estimated.[8] However, since many fabrication plants do not have HEPA filters in the building air exhaust systems, a range of releases, from .108 to 108 kg is estimated. The isotopic composition of the uranium is taken to be that of the product from the enrichment plant, given in Table 4-17.

Likelihood

At least two incidents involving inadvertent releases of UF_6 from cylinder operations have been recorded in fabrication facilities.[31,32] On the basis of the $\sim$490 plant-years of fuel fabrication experience,* the two recorded incidents would result in a UF_6 release likelihood of $\gtrsim 8 \times 10^{-3}$/plant-year.

From the incidents on record at the three enrichment facilities, the likelihood of a release in handling a cylinder was estimated to be approximately 5×10^{-5}/cylinder (see Section 4.3). The generic uranium fabrication plant operating at a nominal capacity of 900 MTU/yr would require the emptying of approximately 600 feed cylinders annually. Then, on the basis of the release likelihood developed from enrichment operations, the likelihood of a release at the fabrication plant is approximately 3×10^{-2}/plant-year.

In another accident evaluation,[45] the probability of a UF_6 cylinder release has been crudely estimated to lie in the range of 10^{-1} to 10^{-3}/year.

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* Assuming, however, that only about one-half of the facilities utilize UF_6 as the feed material.

Estimate: *Source term* $\approx$ *3.5 x* 10^{-2} *to 3.5 x* 10^{-5} *Ci* ^{238}U*, 4.0 x* 10^{-2} *to 4.0 x* 10^{-5} *Ci* ^{237}U*, 7.7 x* 10^{-3} *to 7.7 x* 10^{-6} *Ci* ^{236}U*, 6.0 x* 10^{-3} *to 6.0 x* 10^{-6} *Ci* ^{235}U*, 2.4 x* 10^{-1} *to 2.4 x* 10^{-4} *Ci* ^{234}U *to air; Likelihood* $\approx$ *3 x* 10^{-2}*/plant-year.*

C.2 Failure of Valves or Piping Within the Plant

In the head end of the fuel fabrication process, the vaporized UF_6 is hydrolized to uranyl fluoride and then reacted with ammonia to form ammonium diuranate. Releases of UF_6 occur from time to time due to valve or piping failures, or significant leaks.

Source Term

In an incident on record, a valve on a newly installed unit was inadvertently left open, resulting in approximately 30 kg of uranium as UF_6 released from the process equipment.[32] The magnitude of this release is comparable to the recorded incidents in other components of the fuel cycle. For example, in the recorded incidents at the AEC diffusion plants over the past 20 years, an average of about 44 kg of UF_6 was released within the building.[42]

If a HEPA filter is incorporated within the building ventilation system, a reduction in the amount of uranium released to the environment by 99.9% is estimated. Since, however, the existence of a HEPA filter is not assured, a range of releases, from 440 to 44000 gms, is estimated. The isotopic composition of the uranium is taken to be that of the product from the enrichment plant, given in Table 4-17.

Likelihood

At least one incident involving an inadvertent release of UF_6 from the valves or piping within a fabrication plant has been recorded.[32] On the basis of the $\sim$ 490 plant-years of fuel fabrication experience,* the one incident on record would result in a UF_6 release likelihood of $\overset{\sim}{<}$ 4 x 10^{-3}/ plant-year.

__Estimate:__ Source term $\approx$ *9.7×10^{-3} to 9.7×10^{-6} Ci ^{238}U, 1.1×10^{-2} to 1.1×10^{-5} Ci ^{237}U, 2.1×10^{-3} to 2.1×10^{-6} Ci ^{236}U, 1.7×10^{-3} to 1.7×10^{-6} Ci ^{235}U, 6.8×10^{-2} to 6.8×10^{-5} Ci ^{234}U to air; Likelihood* $\overset{\sim}{>}$ *4×10^{-3}/plant-year.*

D.1 Criticality

Nuclear criticality safety is a consideration in the design, operation and licensing of fuel fabrication plants. Equipment is designed to be maintained in a safe geometry or to contain fixed nuclear poisons to prevent criticality. For the low enrichments involved in LWR fuel processing, the inadvertent formation of a critical mass is a remote possibility. Nevertheless, an incident in which interlocks have failed and a double batch of material has come together can be conceived. The most likely point in the plant for this to occur is at the head end conversion of UF_6 to UO_2, or in scrap recovery, where low enriched uranium is processed in a water solution.

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* Assuming, however that only about one-half of the facilities utilize UF_6 as the feed material.

Source Term

For the ten solution criticalities on record,[31] the average number of fissions was ~ 4×10^{18}. However, one of the incidents involved a record ~ 4×10^{19} fissions at the Chemical Processing Plant of the Idaho Reactor Testing Area, considered unrepresentative of the fuel fabrication operations. Neglecting this incident, the average number of fissions is ~ 4×10^{17}. Other evaluations have selected ~ 10^{18} fissions as representative for criticality in a fuel fabrication plant,[8,16] and this estimate will be adopted here as well.

The radionuclides released ten minutes after an incident involving 10^{18} fissions, based upon ORIGEN calculations,[44] are given in Table 4-21. These estimates further assume 100% of the noble gases and 50% of the halogens released to the environment. Approximately 0.2% of the actinides are assumed released to the building, and if a HEPA filter is present in the building ventilation system, these are attenuated by 99.9%.[8] However, as discussed earlier, a range of particulate releases is adopted, since the existence of a HEPA filter is not assured. Additionally, the neutron and gamma radiation associated with a burst of 10^{18} fissions would result in a dose to the population. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

Likelihood

There have been four criticality incidents associated with fuel fabrication in the ~ 490 plant-years of operations since 1942.[8] From this

TABLE 4-21

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT AT THE URANIUM FUEL FABRICATION FACILITY

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	1.2×10^{-1}	I 136	2.3×10^{1}
Br 80m	5.7×10^{-5}	I 137	3.3×10^{-4}
Br 82	1.9×10^{-5}	Xe 133	1.7×10^{-3}
Br 82m	3.2×10^{-3}	Xe 133m	9.9×10^{-5}
Br 83	3.5	Xe 135	2.2
Br 84	3.5×10^{1}	Xe 135m	4.7
Br 84m	1.5	Xe 137	9.0×10^{2}
Br 85	7.2×10^{1}	Xe 138	7.0×10^{2}
Br 86	1.5	Xe 139	1.5
Br 87	2.1	Xe 140	2.2×10^{-7}
Kr 83m	3.4×10^{-1}	Th 231	2.0×10^{-8} to 2.0×10^{-11}
Kr 85	3.2×10^{-5}	Th 234	2.2×10^{-9} to 2.2×10^{-12}
Kr 85m	1.3×10^{1}	Pa 234m	1.8×10^{-9} to 1.8×10^{-12}
Kr 87	9.4×10^{1}	U 233	1.7×10^{-15} to 1.7×10^{-18}
Kr 88	6.3×10^{1}	U 234	3.4×10^{-13} to 3.4×10^{-16}
Kr 89	5.2×10^{2}	U 235	4.5×10^{-6} to 4.5×10^{-9}
Kr 90	9.2×10^{-2}	U 236	1.3×10^{-11} to 1.3×10^{-14}
I 128	1.4×10^{-3}	U 237	7.1×10^{-5} to 7.1×10^{-8}
I 130	1.0×10^{-3}	U 238	1.1×10^{-5} to 1.1×10^{-8}
I 131	5.5×10^{-2}	U 239	6.1 to 6.1×10^{-3}
I 132	1.6	Np 237	4.4×10^{-16} to 4.4×10^{-19}
I 133	1.5	Np 239	1.5×10^{-2} to 1.5×10^{-5}
I 134	5.2×10^{1}	Np 240	3.5×10^{-11} to 3.5×10^{-14}
I 135	2.3×10^{1}	Pu 239	4.1×10^{-12} to 4.1×10^{-15}

historical data base, a probability of criticality of $\sim 8 \times 10^{-3}$/plant-year is derived for fuel fabrication operations. As discussed in Reference 8, an improved basis for such an estimate would consider the total fuel throughput, the fuel forms during processing, and the fuel reactivities involved.

It is important to note, however, that these four incidents in fabrication operations involved plutonium or fully enriched uranium ($\gtrsim$ 83% enriched in ^{235}U), and that there has not been a criticality incident recorded in any type of operation which involved uranium of low enrichment. Thus the low uranium enrichment associated with LWR fuel is estimated to reduce the likelihood of criticality by at least an order of magnitude.

In another accident evaluation,[45] the probability of a criticality incident in a low enrichment fabrication plant is crudely estimated to lie in the range of 10^{-1} to 10^{-3}/year.

__Estimate:__ Source term - Radionuclides given in Table 4-21 to air plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood $\lesssim 8 \times 10^{-4}$*/plant-year.*

E.1 Waste Retention Pond Failure

The liquid effluents from the UF_6 to UO_2 conversion process and from the scrap recovery operation, which contain strongly acidic or basic chemical wastes and traces of radioactivity, are treated with lime to form a calcium fluoride precipitate, and discharged to a waste retention pond. The pond is sealed to prevent seepage to the ground water. The surface of the pond is exposed to the atmosphere, allowing evaporation to take place. If a dike failure were to occur, or through a natural process, such as earthquake or flooding, several million gallons of contaminated water could be released to the watercourse.

Source Term

At least one incident on record involving a leak in the retention dam of a liquid waste lagoon resulted in a loss of approximately 1.4 million gallons. From the mill tailings pond data (see Section 4.2, Accident E.1), the average release to the watercourse from dike failure or flooding is 3000 m^3 (800,000 gallons).

The concentration of uranium in the liquid waste lagoon is estimated to be approximately 2.5×10^{-5} gm/ml.[45*] Then from the isotopic composition of uranium given in Table 4-17, the estimated concentrations of radionuclides in the waste retention pond are given in Table 4-22.

Likelihood

At least one retention pond leak associated with fuel fabrication activities has been documented,[7] resulting in the loss of approximately

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* This is obtained by scaling both the size of the waste lagoon and the UO_2 production rate from the estimates in Reference 45, and assuming a nominal 4-ft. water depth in the lagoon.

TABLE 4-22

ESTIMATED CONCENTRATIONS OF RADIONUCLIDES IN WASTE RETENTION POND

Isotope	Activity (Ci/ml)
U-238	8.1×10^{-12}
U-237	9.3×10^{-12}
U-236	1.8×10^{-12}
U-235	1.4×10^{-12}
U-234	5.7×10^{-11}
Th-234	8.1×10^{-12}
Pa-234	8.1×10^{-12}

1.4 million gallons of contaminated liquids. On the basis of 490 plant-years of fuel fabrication activities, this would result in a release likelihood of $\sim 2 \times 10^{-3}$/plant-year.

From the mill tailings pond data (see Section 4.2, Accident E.1), the likelihood of a release involving the source term described in the previous section is approximately 2×10^{-2}/plant-year of mill operation. As pointed out in Section 4.2, however, this estimate may be high, since the evaluation of early dike construction led to strengthening of existing dikes and improved construction standards.

Estimate: *Source term* $\approx 2.4 \times 10^{-2}$ *Ci* ^{238}U, 2.8×10^{-2} *Ci* ^{237}U, 5.4×10^{-3} *Ci* ^{236}U, 4.2×10^{-3} *Ci* ^{235}U, 1.7×10^{-1} *Ci* ^{234}U, 2.4×10^{-2} *Ci* ^{234}Th, 2.4×10^{-2} *Ci* ^{234}Pa *to watercourse; Likelihood* $\approx 2 \times 10^{-2}$ *to* 2×10^{-3}*/plant-year.*

4.6 Reprocessing

The accidents considered in spent fuel reprocessing facilities, keyed to the accident categories given in Table 4-1, are listed in Table 4-23.

Other accidents can be postulated to occur in reprocessing operations. For example, fission gases could be released from the fuel pins should the fuel cask be dropped during unloading or if the fuel element becomes overheated during transfer to the shear operation. However, the risk associated with these events is judged to be small in comparison with the accident examined in the fuel receiving and storage area. Fires involving leached zirconium hulls have occurred, but the duration, in general, is short, and the risk is judged to be insignificant in comparison with the fires considered in Table 4-23. Small leaks in vessels or pipes containing radioactive material, or operator errors resulting in inadvertent discharges of radioactive solutions may occur relatively frequently,[32] but the source term to the environment is generally negligible in comparison with the accidents analyzed in this section. Similarly, filter failures or loss of ventilation zone differential pressure may occur, but the small releases which result during the abnormal condition are generally an insignificant fraction of the normal annual release.

High level wastes are stored in solution for up to five years prior to solidification. The radiolytic heat generated by the fission products requires water cooling of the high level waste tanks. A loss of coolant

TABLE 4-23

SPENT FUEL REPROCESSING ACCIDENTS

A.1	Explosion in the high aqueous waste concentrator
A.2	Explosion in the low aqueous waste concentrator
A.3	Explosion in the high aqueous feed tank
A.4	Explosion in the waste calciner
A.5	Explosion in the iodine adsorber
B.1	Solvent fire in the codecontamination cycle
B.2	Solvent fire in the plutonium extraction cycle
B.3	Ion-exchange resin fire
C.1	Fuel assembly rupture and release in fuel receiving and storage area
C.2	Dissolver seal failure
C.3	Release from a hot UF_6 cylinder
D.1	Criticality

to these tanks could result in the release of a significant amount of activity, but is only remotely possible because of the defenses in depth which would be operative. The risk associated with this accident was not considered in this study due to a lack of pertinent data.

The potentially toxic effects resulting from the release of acids or hydrogen fluoride to the environment is considered outside of the scope of the current study.

Current regulatory criteria related to the design of a reprocessing facility require that the structures, systems and components withstand the effects of natural phenomena. These include all floods, tornados, earthquakes, or missiles of intensity more severe than experienced historically in the locality of the plant. Although events with intensities outside of this range are conceivable, the exceedingly low probabilities associated with their occurrence were not evaluated in the current study.

Only one commercial reprocessing facility has operated in the United States. This plant, which is located in West Valley, New York, was operated by Nuclear Fuel Services in the period from 1966 to 1972. Federally supported facilities which employ processes similar to current designs for LWR spent fuel reprocessing plants, are in operation at Hanford, Savannah River, Oak Ridge, and Idaho. If these facilities are included, approximately 100 plant-years of experience have been accumulated in the reprocessing of spent fuel.

Most of the data presented in this section were developed in an earlier hazards analysis of a generic fuel reprocessing facility.[9] The Safety Analysis Report for the Barnwell Nuclear Fuel Plant[46] was the source of most of the release data presented in that study.

A.1 Explosion in the High Aqueous Waste (HAW) Concentrator

The high aqueous waste concentrator, located in the remote process cell, concentrates the high-level radioactive waste streams from all the solvent extraction cycles to recover nitric acid and water for reuse in the process, while reducing the waste volumes for storage in waste facilities. An explosion in the HAW waste concentrator could conceivably be caused by ignition of an explosive mixture of hydrogen in the air above the liquid in the evaporator or a "red-oil" explosion. Hydrogen and oxygen are generated by radiolysis of aqueous solutions. To avoid reaching a combustible hydrogen concentration, dilution of the off-gases with continously flowing air is used in the processing operations. However, a failure of the air purge system through failure of blowers or their power supply, filter blockages, or ventilation control failure could result in a hydrogen explosion. To reduce the likelihood of air flow failure, the plant is designed with redundant air flow features.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a HAW concentrator explosion are given in Table 4-24.[9] These values are based upon an estimated concentrator volume of 600

TABLE 4-24

RADIONUCLIDE RELEASE RESULTING FROM AN HAW CONCENTRATOR EXPLOSION AT THE REPROCESSING FACILITY[9]

Nuclide	Activity in Fuel (Ci/MTHM)	Activity Released* (Ci)
Sr 89	9.0×10^4	3.1×10^{-4}
Sr 90	8.4×10^4	2.9×10^{-4}
Y 90	8.4×10^4	2.9×10^{-4}
Y 91	1.9×10^5	6.6×10^{-4}
Zr 95	3.5×10^5	1.2×10^{-3}
Nb 95	6.5×10^5	2.3×10^{-3}
Ru 103	1.2×10^5	1.3×10^2
Ru 106	6.1×10^5	6.4×10^2
I 129	3.6×10^{-2}	3.1×10^{-3}
I 131	1.6	1.4×10^{-1}
Cs 134	2.4×10^5	8.3×10^{-4}
Cs 137	1.2×10^5	4.2×10^{-4}
Ce 141	7.9×10^4	2.7×10^{-4}
Ce 144	8.8×10^5	3.1×10^{-3}
Pm 147	1.4×10^5	4.8×10^{-4}
Pu 238	4.3×10^3	1.5×10^{-5}
Pu 239	3.2×10^2	1.1×10^{-6}
Pu 240	6.3×10^2	2.2×10^{-6}
Pu 241	1.7×10^5	5.9×10^{-4}
Pu 242	3.6	1.3×10^{-8}
Am 241	2.5×10^2	8.7×10^{-7}
Am 242	4.0	1.4×10^{-8}
Cm 242	4.4×10^4	1.5×10^{-4}
Cm 243	3.4×10^1	1.2×10^{-8}
Cm 244	5.7×10^3	2.0×10^{-6}

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* The values tabulated are for normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 10^5. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part

liters with an equivalence of 1.76 kg of heavy metal per liter. The estimated fraction of radionuclides in the concentrate which were originally present in the fuel are 8.3×10^{-2} for iodine, 10^{-2} for plutonium, and unity for the remainder. The estimated volatile fraction of ruthenium is 10^{-3} and of iodine is unity. The amount of concentrate released to the cell is limited by the allowable concentration of respirable heavy particles in the air, approximately 100 mg/m^3.[8] The remote process cell volume is taken to be 2850 m^3. The estimated fraction of non-volatile material passing through two HEPA filters in series under normal conditions is 10^{-5}.[8] In the event of simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

Likelihood

The likelihood of an HAW concentrator explosion in a generic spent fuel reprocessing facility with normal HEPA filter operation has been estimated to be approximately 10^{-5}/yr.[9] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

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* vapor and part particulate. In the event of simultaneous filter failure, the ruthenium activity released is assumed to be the following: ^{103}Ru - 1.7×10^2 Ci, ^{106}Ru - 8.5×10^2 Ci.

Estimate: Source term: Radionuclide given in Table 4-24 to air for normal filter operation and the event of filter failure; Likelihood $\approx 10^{-5}$*/year for normal filter operation,* 10^{-8}*/year for simultaneous failure of two series HEPA filters.*

A.2 Explosion in the Low Activity Waste (LAW) Concentrator

The low activity waste concentrator, located in the high intermediate level cell, concentrates the low-level radioactive waste streams from all the solvent extraction cycles. An explosion in the LAW concentrator could also be caused by a hydrogen explosion or a "red-oil" explosion. "Red-oil" is a material that can be formed from a heavy metal nitrate and/or nitric acid solutions mixed with tributyl phosphate solvent at temperatures exceeding 135°C. Under optimum conditions, the reaction is explosive and oxides of nitrogen are evolved. In order for a "red-oil" explosion to occur, several independent instrument control failures which are designed to keep the temperature in the waste concentrators below 135°C and to keep the solvent out of the aqueous stream would have to occur.

Source Term

The estimated quantities of radionuclides released to the environment in the event of an LAW concentrator explosion are given in Table 4-25.[9] These values are based upon an estimated concentrator volume of 1500 liters with an equivalence of 2.31 kg of heavy metal per liter. The estimated fractions of radionuclides in the concentrate which were

TABLE 4-25

RADIONUCLIDE RELEASE RESULTING FROM A LAW CONCENTRATOR EXPLOSION AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Sr 89	5.0×10^{-7}
Sr 90	4.7×10^{-7}
Y 90	4.7×10^{-7}
Y 91	1.1×10^{-6}
Zr 95	2.0×10^{-5}
Nb 95	3.6×10^{-5}
Ru 103	8.4
Ru 106	4.3×10^{1}
I 129	4.0×10^{-3}
I 131	1.8×10^{-1}
Cs 134	1.3×10^{-6}
Cs 137	6.7×10^{-7}
Ce 141	4.4×10^{-7}
Ce 144	4.8×10^{-6}
Pm 147	7.8×10^{-7}
Pu 238	1.1×10^{-8}
Pu 239	8.1×10^{-10}
Pu 240	8.1×10^{-10}
Pu 241	4.3×10^{-7}
Pu 242	9.1×10^{-12}
Am 241	1.4×10^{-9}
Am 242	2.2×10^{-11}
Cm 242	2.5×10^{-7}
Cm 243	1.9×10^{-10}
Cm 244	3.2×10^{-8}

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* The values tabulated are for normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 10^5. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part vapor and part particulate. In the event of simultaneous filter failure, the ruthenium activity released is assumed to be the following: ^{103}Ru - 9.1 Ci, ^{106}Ru - 4.6×10^{1} Ci.

originally present in the fuel are 3.2 x 10^{-2} for iodine, 9 x 10^{-4} for plutonium, 2 x 10^{-2} for ruthenium, zirconium and niobium, and 2 x 10^{-3} for the remainder. The estimated volatile fraction of ruthenium is 10^{-3} and of iodine in unity. The amount of concentrate released to the cell is limited by the allowable concentration of respirable heavy particles in the air, approximately 100 mg/m^3.[8] The high intermediate level cell volume is taken to be 1510 m^3. The estimated fraction of non-volatile material passing through the two HEPA filters in series under normal conditions is 10^{-5}.[8] In the event of a simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

Likelihood

The likelihood of a LAW concentrator explosion in a generic spent fuel reprocessing facility with normal filter operation has been estimated to be approximately 10^{-4}/year.[9] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term: Radionuclides given in Table 4-25 to air for normal filter operation and in the event of filter failure; Likelihood ≈ *10^{-4}/year for normal filter operation, 10^{-7}/year for simultaneous failure of two series HEPA filters.*

A.3 Explosion in the High Aqueous Feed (HAF) Tank

The high aqueous feed tank, located in the high level cell, receives the solution from the dissolution step and feeds the extraction steps of the process. An explosion in the HAF tank could be caused by ignition of an explosive mixture of radiolytically generated hydrogen in the air above the liquid. The occurrence of an explosive concentration of hydrogen would require a failure in the air purge system.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a HAF tank explosion are given in Table 4-26.[9] These values are based upon estimated fractions of radionuclides in the solution which were originally present in the fuel of 0.5 for iodine and unity for all the remaining radionuclides. The estimated volatile fraction of ruthenium is 10^{-3} and of iodine is unity. The estimated fraction of the non-volatile radionuclides which is dispersed to the cell from the explosion is 5.9×10^{-6}. The non-volatile radionuclides are assumed to pass through one HEPA filter which is estimated to trap 99.9% of the respirable material.[8] In the event of a filter failure, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

Likelihood

The likelihood of a HAF tank explosion in a generic fuel reprocessing facility with normal HEPA filter operation has been estimated to be

TABLE 4-26

RADIONUCLIDE RELEASE RESULTING FROM A HAF TANK EXPLOSION AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Sr 89	2.1×10^{-3}
Sr 90	2.0×10^{-3}
Y 90	2.0×10^{-3}
Y 91	4.5×10^{-3}
Zr 95	8.2×10^{-3}
Nb 95	1.5×10^{-2}
Ru 103	4.8×10^{2}
Ru 106	2.4×10^{3}
I 129	7.2×10^{-2}
I 131	3.2
Cs 134	5.7×10^{-3}
Cs 137	2.8×10^{-3}
Ce 141	1.9×10^{-3}
Ce 144	2.1×10^{-2}
Pm 147	3.3×10^{-3}
Pu 238	1.0×10^{-4}
Pu 239	7.6×10^{-6}
Pu 240	1.5×10^{-5}
Pu 241	4.0×10^{-3}
Pu 242	8.5×10^{-8}
Am 241	5.9×10^{-3}
Am 242	9.5×10^{-8}
Cm 242	1.0×10^{-3}
Cm 243	8.0×10^{-7}
Cm 244	1.4×10^{-4}

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* The values tabulated are for normal operation of the single HEPA filter. In the event of failure of the filter, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 10^3. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part vapor and part particulate. In the event of filter failure, however, the ruthenium activity released is unchanged.

approximately 10^{-5}/year.[9] The likelihood of a filter failure has been estimated to be roughly 10^{-2}/demand.

Estimate: Source term: Radionuclides given in Table 4-26 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 10^{-5}$*/year for normal filter operation,* 10^{-7}*/year for filter failure.*

A.4 Explosion in the Waste Calciner

The concentrated wastes are fed from the HAW concentrator to the waste calciner, where the high level wastes are calcined at a temperature in excess of 450°C. The waste calciner is located in the remote process cell. An explosion in the waste calciner could be initiated by ignition of an explosive mixture of hydrogen in air or by an excess pressure buildup in the steam supply. In either case, several independent instrument control failures would have to occur.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a waste calciner explosion are given in Table 4-27.[9] The source term for this accident is identical to that of the HAW concentrator explosion, with one exception. Since the calciner operates at several hundred degrees centigrade, the amount of ruthenium volatilized in the course of the accident is estimated to be approximately a factor of ten higher than that volatilized in the course of the HAW concentrator explosion.

TABLE 4-27

RADIONUCLIDE RELEASE RESULTING FROM AN EXPLOSION IN THE WASTE CALCINER AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Sr 89	3.1×10^{-4}
Sr 90	2.9×10^{-4}
Y 90	2.9×10^{-4}
Y 91	6.6×10^{-4}
Zr 95	1.2×10^{-3}
Nb 95	2.3×10^{-3}
Ru 103	1.3×10^{3}
Ru 106	6.4×10^{3}
I 129	3.1×10^{-3}
I 131	1.4×10^{-1}
Cs 134	8.3×10^{-4}
Cs 137	4.2×10^{-4}
Ce 141	2.7×10^{-4}
Ce 144	3.1×10^{-3}
Pm 147	4.8×10^{-4}
Pu 238	1.5×10^{-5}
Pu 239	1.1×10^{-6}
Pu 240	2.2×10^{-6}
Pu 241	5.9×10^{-4}
Pu 242	1.3×10^{-8}
Am 241	8.7×10^{-7}
Am 242	1.4×10^{-8}
Cm 242	1.5×10^{-4}
Cm 243	1.2×10^{-8}
Cm 244	2.0×10^{-6}

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* The values tabulated are for normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 10^5. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part vapor and part particulate. In the event of simultaneous filter failure, the ruthenium activity released is assumed to be the following: ^{103}Ru - 1.4×10^3 Ci, ^{106}Ru - 6.6×10^3 Ci.

Likelihood

The likelihood of a waste calciner explosion in the generic fuel reprocessing facility with normal HEPA filter operation has been estimated to be approximately 10^{-6}/year.[9] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term: Radionuclides given in Table 4-27 to air for normal filter operation and in the event of filter failure; likelihood $\approx 10^{-6}$/year for normal filter operation, 10^{-9}/year for simultaneous failure of two series HEPA filters.

A.5 Explosion in the Iodine Adsorber

Silver zeolite sorbents are incorporated in the process off-gas treatment to limit the release of radioiodine to the environment. A silver reactor explosion could result from the introduction of ammonia to the sorbent with resultant formation of an explosive azide compound. Although ammonia is not used in the process, the inadvertent use of plant reagent chemicals such as hydrazine or hydroxylamine nitrate, which are used in the uranium-plutonium partitioning and extraction steps, could result in the formation of ammonia vapors.

Source Term

Assuming that the radioiodine present in 4 MT of partially dissolved fuel were available for release to the atmosphere, and that one-half of the volatilized radioiodine is removed by the exhaust filter, the estimated release to the environment of I-131 is 6 Ci.[9] Ratioing the activities of I-129 and I-131 in the fuel, the estimated release of I-129 is approximately 1.4×10^{-1} Ci.

Likelihood

The likelihood of an iodine adsorber explosion in the generic fuel reprocessing facility has been estimated to be approximately 2×10^{-4}/year.[9]

Estimate: Source term: 6 Ci of ^{131}I *and* 1.4×10^{-1} *Ci of* ^{129}I *to air; Likelihood* $\approx 2 \times 10^{-4}$*/year.*

B.1 Solvent Fire in the Codecontamination Cycle

Codecontamination is the operation which removes most of the fission products and other undesirable impurities from solutions of uranium and plutonium without separating the uranium and plutonium components. The solvent extraction cycle, which takes place in the high intermediate level cell, employs 30 volume percent tributyl phosphate in a normal parafinic hydrocarbon (dodecane) solvent. Because of the relatively low flash point ($\sim$70°C) of the organic solvent, there exists the potential for a solvent fire during this processing step. The operating temperature is maintained below 70°C by temperature controls and the flow rates are monitored to avoid spills and to maintain the desired compositions in all feed and discharge streams of the equipment used. A solvent fire could result from the failure of temperature control which would allow the flash point to be reached.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a solvent fire in the codecontamination cycle are given in Table 4-28.[9] These releases are based upon an estimated 378 liters of solvent burned with an equivalence of 0.23 kg heavy metal per liter. The estimated fraction of the radionuclides in the solvent which were originally present in the fuel are 4×10^{-2} for iodine, 10^{-2} for plutonium, ruthenium, zirconium and niobium, and 1×10^{-3} for the remainder. The estimated volatile fraction of ruthenium is 10^{-1} and of iodine is unity. It is assumed that 1% of the non-volatile radionuclides in the burned solvent are dispersed by the fire. The estimated fraction of the dispersed, non-volatile material passing through the two HEPA filters under normal conditions in series is 10^{-5}.[8] In the event of simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

Likelihood

The likelihood of a solvent fire in the codecontamination cycle of a generic fuel reprocessing facility has been estimated to fall in the range of 10^{-5} to 10^{-6}/year.[9] However, from chemical industry data, the probability of major fires per plant-year is estimated to be 4×10^{-4}.[8] This probability is reduced to 2×10^{-4}/plant-year in Reference 8 for a generic mixed oxide fuel fabrication plant, in which defenses are employed uncharacteristic of the typical chemical plant. The likelihood

TABLE 4-28

RADIONUCLIDE RELEASE RESULTING FROM A FIRE IN THE CODECONTAMINATION CYCLE AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Sr 89	1.5×10^{-7}
Sr 90	7.1×10^{-7}
Y 90	7.1×10^{-7}
Y 91	3.7×10^{-7}
Zr 95	4.2×10^{-6}
Nb 95	7.7×10^{-6}
Ru 103	1.6
Ru 106	3.8×10^{1}
I 129	1.2×10^{-4}
I 131	1.7×10^{-4}
Cs 134	1.6×10^{-6}
Cs 137	9.8×10^{-7}
Ce 141	8.7×10^{-8}
Ce 144	4.9×10^{-6}
Pm 147	1.1×10^{-6}
Pu 238	3.7×10^{-7}
Pu 239	2.7×10^{-8}
Pu 240	5.4×10^{-8}
Pu 241	1.4×10^{-5}
Pu 242	3.1×10^{-10}
Am 241	2.1×10^{-9}
Am 242	3.4×10^{-11}
Cm 242	1.6×10^{-7}
Cm 243	2.9×10^{-10}
Cm 244	4.8×10^{-8}

---

* The values tabulated are for normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 10^5. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part vapor and part particulate. In the event of simultaneous filter failure, the ruthenium activity released is assumed to be the following: ^{103}Ru - 1.8 Ci, ^{106}Ru - 4.2×10^{1} Ci.

of a simultaneous failure of the two series HEPA filters has been estimated to be roughly 10^{-3}/demand.[9]

__Estimate:__ Source term: Radionuclides given in Table 4-28 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 10^{-4}$ *to* 10^{-6}*/plant-year for normal filter operation,* 10^{-7} *to* 10^{-9}*/plant-year for simultaneous failure of the two series HEPA filters.*

B.2 Solvent Fire in the Plutonium Extraction Cycle

The organic solution from the codecontamination step may be passed through a partitioning column located in the plutonium product cell, where tetravalent plutonium is electrochemically reduced to the less extractable trivalent state, and subsequently stripped into another aqueous nitric acid stream containing hydrazine.* Solvent extraction cycles are also used for the partitioning operation in conjunction with various chemical adjustments. As in the codecontamination step, a solvent fire is possible in the partitioning process, and is avoided by limiting the operating temperature of the process.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a solvent fire in the plutonium extraction cycle are given in Table 4-29.[9] Radionuclides other than plutonium are considered negligible. These releases are based upon an estimated 14 liters of solvent burned with an equivalence of 2.44 kg heavy metal per liter. The estimated fraction of plutonium in the solvent which was originally present in the fuel is unity. It is assumed that 1% of the plutonium in the burned solvent is dispersed by the fire. The estimated fraction of the dispersed plutonium passing through the three HEPA filters in series under normal conditions is 6×10^{-7}.[8] In the event of a simultaneous failure of the three HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

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* Alternatively, anion exchange could be used for partitioning plutonium and uranium into separate streams.

Likelihood

The likelihood of a solvent fire in the plutonium extraction cycle of a generic fuel reprocessing facility has been estimated to fall in the range of 10^{-5} to 10^{-6}/year.[9] However, from chemical industry data, the probability of major fires per plant-year is estimated to be 4×10^{-4}.[8] This probability is reduced to 2×10^{-4}/plant-year in Reference 8 for a generic mixed oxide fuel fabrication plant, in which defenses are employed uncharacteristic of the typical chemical plant.

The failure of a single HEPA filter has been estimated to be 10^{-2}/event, whereas the failure of two HEPA filters in series has been estimated to be 10^{-3}/event.[9] Since the cell exhaust filter and the exit filters from the ventilation system are assumed to be independent, the combined probability of all three filters failing simultaneously is further estimated to be 10^{-5}/demand.

TABLE 4-29

RADIONUCLIDE RELEASE RESULTING FROM A FIRE IN THE PLUTONIUM EXTRACTION CYCLE AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Pu 238	8.8×10^{-7}
Pu 239	6.5×10^{-8}
Pu 240	1.3×10^{-7}
Pu 241	3.5×10^{-5}
Pu 242	7.3×10^{-10}

---

* The values tabulated are for normal operation of the three series HEPA filters. In the event of simultaneous failure of the filters, the values in this column are increased by a factor of 1.6×10^6.

Estimate: Source term: Radionuclides given in Table 4-29 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 10^{-4}$ *to* 10^{-6}*/plant-year for normal operation,* 10^{-9} *to* 10^{-11}*/plant-year for simultaneous failure of three series HEPA filters.*

B.3 Ion-Exchange Resin Fire

Ion exchange resin columns, contained in the plutonium product cell, are used to partition plutonium, uranium and/or neptunium into separate streams, while providing for additional fission product decontamination following the initial codecontamination step. A resin-nitrate reaction in these columns could lead to an ion exchange resin fire. A fire is prevented by process control which limits the acidity and temperature of the resin to less than 135°C. If the resin beds are highly loaded with plutonium, radiolytic heating could augment the temperature rise of the bed.

Source Term

The estimated quantities of radionuclides released to the environment in the event of an ion-exchange resin fire are given in Table 4-30.[9] The estimated fractions of radionuclides in the resin which were originally present in the fuel are 3.0×10^{-6} for ruthenium, 5.0×10^{-4} for plutonium, 0.95 for neptunium, 6.6×10^{-6} for zirconium and niobium, 3.1×10^{-7} for iodine, and 1.0×10^{-8} for the remainder. In this estimate, the volatile fraction of ruthenium is taken to be 5×10^{-2} and of iodine, 0.5. All of the activity contained within the resin from a full day's processing is assumed to be released to the cell in the fire. The

TABLE 4-30

RADIONUCLIDE RELEASE RESULTING FROM AN ION-EXCHANGE RESIN FIRE AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Released (Ci)*
Sr 89	1.2×10^{-9}
Sr 90	1.2×10^{-9}
Zr 95	3.5×10^{-6}
Nb 95	6.8×10^{-6}
Ru 103	9.3×10^{-2}
Ru 106	5.4×10^{-1}
I 129	2.8×10^{-8}
I 131	1.2×10^{-6}
Cs 134	2.7×10^{-9}
Cs 137	1.9×10^{-9}
Ba 137m	1.7×10^{-9}
Ce 144	8.4×10^{-9}
Np 238	9.1×10^{-6}
Pu 238	3.3×10^{-6}
Pu 239	2.4×10^{-7}
Pu 240	4.4×10^{-7}
Pu 241	1.2×10^{-4}
Cm 242	6.0×10^{-10}
Cm 244	7.1×10^{-11}

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* The values tabulated are for normal operation of the three series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine and ruthenium, are increased by a factor of 1.6×10^6. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged. Ruthenium is part vapor and part particulate. In the event of simultaneous filter failure, the ruthenium activity released is assumed to be the following: ^{103}Ru - 9.3×10^{-1} Ci, ^{106}Ru - 5.4 Ci.

estimated fraction of non-volatile material passed through the three HEPA filters in series under normal conditions is 6×10^{-7}.[8] In the event of simultaneous failure of the three HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere.

Likelihood

The likelihood of an ion exchange resin fire in a generic fuel reprocessing facility has been estimated to be approximately 10^{-4}/year.[9] However, an estimate of the likelihood of an ion exchange fire in a mixed oxide fuel fabrication plant has been given as $< 10^{-1}$/year.[8]

Four incidents have been reported resulting in a release of radioactivity as a result of thermochemical instabilities in ion-exchange processing.[8] On the basis of 490 plant-years of fuel fabrication activities, this history would result in a likelihood of $\sim 8 \times 10^{-3}$/plant-year.

At least one incident has been reported involving a fire around an anion exchange column in fuel reprocessing activities (at Savannah River).[31] On the basis of 100 plant-years of reprocessing activities, this history would result in a likelihood of $\sim 10^{-2}$/plant year.

The failure of a single HEPA filter has been estimated to be 10^{-2}/event, whereas the failure of two HEPA filters in series has been estimated to be 10^{-3}/event.[9] Since the cell exhaust filter and the exit filters from the ventilation system are assumed to be independent, the combined probability of all three filters failing simultaneously is further estimated to be 10^{-5}/demand.

Estimate: Source term: Radionuclides given in Table 4-30 to air for normal filter operation and in the event of filter failure; Likelihood ≈ 10^{-1} to 10^{-4}/plant-year for normal filter operation, 10^{-6} to 10^{-9}/plant-year for simultaneous failure of three series HEPA filters.

C.1 Fuel Assembly Rupture and Release in Fuel Receiving and Storage Area

Irradiated fuel assemblies arrive at the reprocessing plant in shielded casks, where they are removed from the carriers and submerged in a pool of water for unloading the fuel assemblies. The cask is opened and the fuel assemblies removed and placed in storage canisters. If the cask were to lose its heat removal capability during shipment, the self-heating of the spent fuel rods from fission products could elevate the cladding temperature beyond the failure point. On opening the cask, mobile radionuclides would be expelled from the cask cavity as a stream of bubbles which rise to the pool surface.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a fuel assembly rupture and release in the fuel receiving and storage area are given in Table 4-31.[9] This estimate is based upon the assumption that all of the noble gases, tritium and radioiodine; 10% of the ruthenium; and 1% of the cesium and strontium in the breached elements are released to the pool water. The airborne release of noble gases and tritium are neglected in the accident evaluation since they

TABLE 4-31

RADIONUCLIDE RELEASE RESULTING FROM A FUEL ASSEMBLY RUPTURE AND RELEASE IN FUEL RECEIVING AND STORAGE AREA AT THE REPROCESSING FACILITY[9]

Nuclide	Activity Release (Ci)*
Ru 103	3.8×10^{-3}
Ru 106	1.9×10^{-2}
I 129	1.6×10^{-11}
I 131	7.2×10^{-10}
Cs 134	1.1×10^{-7}
Cs 137	5.4×10^{-8}

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* The values given assume normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column are increased by a factor of 10^5.

are normally released in the dissolution step of the process. Ninety percent of the radioiodine and ruthenium; 99.9% of the cesium; and all of the strontium are assumed to remain in the pool water. The released gases subsequently pass through a scrubber, which removes 93% of the ruthenium, 99.9% of the cesium, and 99.99% of the iodine, and two HEPA filters in series, which under normal conditions pass 10^{-5} of the materials.* In the event of simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the fuel receiving and storage area is assumed to be released to the atmosphere.

Likelihood

The likelihood of a fuel assembly rupture and release in the fuel receiving and storage area has been estimated to be in the range of 10^{-1} to 10^{-2}/year.[9] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9] The potential failure of the scrubber has not been considered, since the failure probability would have to be in excess of $\sim 10^{-1}$/event before a significant contribution to risk would result.

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* It is assumed in Reference 9 that the materials released from the cask as vapors are converted to the particulate form in passage through the pool.

Estimate: Source term - Radionuclides given in Table 4-31 to air for normal filter operation and in the event of filter failure; Likelihood $\approx$ *10^{-1} to 10^{-2}/plant-year for normal filter operation, 10^{-4} to 10^{-5}/plant-year for simultaneous failure of two series HEPA filters.*

C.2 Dissolver Seal Failure

The segmented fuel containing the unspent uranium and radionuclides formed during irradiation is dissolved out of the cladding hulls with nitric acid to form the feed for subsequent extraction steps. It is assumed that a leak in the seam of the dissolver releases the solution onto a hot surface, producing a powder of all the contained non-volatile materials and evolving the volatile materials.

Source Term

The estimated quantities of radionuclides released to the environment in the event of a dissolver seal failure is given in Table 4-32.[42] These estimates are based upon the assumption that all materials associated with one kg of heavy metal are dispersed to the cell atmosphere, and subsequently pass through the cell ventilation system.

Likelihood

An estimate in the range of 10^{-5} to 10^{-6}/year has been made of the likelihood of plumbing failure due to corrosion in the high-level and intermediate-level cells of a generic fuel reprocessing plant.[9] In the same source, an estimate of 10^{-5}/year has been made for the likelihood of failure in the primary boundary. The likelihood of a

TABLE 4-32

RADIONUCLIDE RELEASE RESULTING FROM A DISSOLVER SEAL FAILURE AT THE REPROCESSING FACILITY[42]

Nuclide	Activity Released (Ci)*
Sr 89	6.52×10^{-6}
Sr 90	4.67×10^{-5}
Y 90	4.67×10^{-5}
Y 91	1.64×10^{-5}
Zr 95	4.57×10^{-5}
Nb 95	9.71×10^{-5}
Ru 103	4.16×10^{-6}
Ru 106	6.88×10^{-4}
Ag 110	3.93×10^{-7}
Sb 125	2.03×10^{-5}
Te 127	3.54×10^{-6}
Te 129	4.20×10^{-8}
I 129	3.92×10^{-6}
I 131	3.10×10^{-12}
Cs 134	2.11×10^{-5}
Cs 137	1.24×10^{-4}
Ce 141	7.46×10^{-7}
Ce 144	5.18×10^{-4}
Pm 147	2.71×10^{-4}
Eu 154	1.44×10^{-6}
Eu 155	3.78×10^{-5}
U 234	1.66×10^{-10}
U 235	2.64×10^{-12}
U 236	8.24×10^{-12}
U 238	2.87×10^{-10}
Pu 238	1.89×10^{-5}
Pu 239	3.85×10^{-6}

TABLE 4-32

(continued)

Nuclide	Activity Released (Ci)*
Pu 240	5.24×10^{-6}
Pu 241	5.71×10^{-4}
Am 241	2.95×10^{-6}
Am 243	5.41×10^{-8}
Cm 242	1.73×10^{-5}
Cm 244	1.44×10^{-6}

---

* The values calculated are for normal operation of the two series HEPA filters. In the event of simultaneous failure of the filters, the values in this column, with the exception of iodine, are increased by a factor of 10^5. Since iodine is assumed to be a vapor, the activity released is assumed to be unchanged.

simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

__Estimate:__ Source Term: Radionuclides given in Table 4-32 to air for normal filter operation and in the event of filter failure; Likelihood $\approx$ *10^{-5}/plant-year for normal filter operation, 10^{-8}/plant-year for simultaneous failure of two series HEPA filters.*

C.3 Release from a Failure of Hot UF_6 Cylinder

The recovered uranium from fuel reprocessing is converted to UF_6 for shipment to the enrichment plant. The transfer of UF_6 from the surge tanks to the cylinders for shipment is accomplished by melting the UF_6 into a pressurized liquid. The cylinder is removed from the loadout area by fork lift to an outdoor storage area where the UF_6 cools and solidifies. At any time during this sequence of events, while the UF_6 is in the liquid state, the failure of a valve, an operator error, or a cylinder rupture could release significant quantities of UF_6 to the environment.

Source Term

A summary of releases associated with hookup and disconnect operations on UF_6 cylinders at the three existing gaseous diffusion plants has been compiled and was summarized in Table 4-12. The average release in six recorded incidents resulting in more than 5 kg loss was 108 kg of uranium. Since there exists no data base associated with UF_6 releases at reprocessing plants, the average source term from enrichment will be adopted here. The specific activities by isotope of uranium having undergone 33,000 MWD/MT burnup are given in Table 4-33.[47]

Likelihood

From the incidents on record at the three enrichment facilities, the likelihood of a release in handling a cylinder was estimated to be approximately 5 x 10^{-5}/cylinder (see Section 4.3). The generic reprocessing fabrication plant operating at a nominal capacity of 1500 MT HM/year would require the filling of approximately 1000 cylinders annually of 2.2 MT UF_6 capacity. Thus, on the basis of the release likelihood developed from enrichment operations, the likelihood of a release at the reprocessing facility is approximately 5 x 10^{-2}/plant-year.

Estimate: *Source term* $\approx$ *3.6 x* 10^{-2} *Ci* ^{238}U*, 3.0 x* 10^{-1} *Ci* ^{237}U*, 3.2 x* 10^{-2} *Ci* ^{236}U*, 1.9 x* 10^{-3} *Ci* ^{235}U*, 8.5 x* 10^{-2} *Ci* ^{234}U*, 3.6 x* 10^{-2} *Ci* ^{234}Th *to air; Likelihood* $\approx$ *5 x* 10^{-2}*/plant-year.*

D.1 Criticality

Accidental criticality in fuel receiving and storage operations is unlikely because the areas where these operations are performed are designed to be subcritical with unirradiated fuel of 5% enrichment. Light water reactor fuel is normally enriched to less than 4%, and after burnup the enrichment is significantly reduced and fission product poisons are present in the fuel.

TABLE 4-33

ISOTOPIC ACTIVITIES OF URANIUM WITH 33,000 MWD/MT BURNUP[47]

Isotope	Activity per Gram of Uranium (Ci)
^{238}U	3.3×10^{-7}
^{237}U	2.8×10^{-6}
^{236}U	3.0×10^{-7}
^{235}U	1.8×10^{-8}
^{234}U	7.9×10^{-7}
^{234}Th	3.3×10^{-7}

Criticality could accidentally occur by overfilling a dissolver, or by accidental transfer of plutonium fines to the dissolver transfer tank and to the accountability tank. Following the concentration of uranium and plutonium, criticality is inhibited by controlling the concentrations of fissile materials in the solutions. A criticality accident could occur in this part of the process from a failure of process control that results in higher-than-normal fissile material concentrations in solution concurrently with multiple monitoring failures or it could result from administrative error by processing higher enrichment fuel under specifications normally used for lower uranium enrichment.

Criticality in product loadout is also possible, particularly in the plutonium loadout area, if failure of both concentration control and monitor failure were to occur, or if the plutonium product cell were to be flooded.

Source Term

As discussed in Section 4.5 (accident D.1), 10^{18} fissions are selected as representative of the energy release in a solution criticality for purposes of this study.

The radionuclides released to the environment ten minutes after an inadvertent criticality involving plutonium and resulting in 10^{18} fissions, based upon ORIGEN calculations (for a ^{235}U criticality incident)[44] are given in Table 4-34. These estimates assume that 100% of the noble gases are released to the environment, but that prior to release, 99.99% of the halogens are removed by scrubbers. Approximately 0.2% of the

TABLE 4-34

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT AT THE FUEL REPROCESSING FACILITY

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	2.4×10^{-5}	I 136	4.6×10^{-3}
Br 80m	1.1×10^{-8}	I 137	6.6×10^{-8}
Br 82	3.8×10^{-9}	Xe 133	1.7×10^{-3}
Br 82m	6.4×10^{-7}	Xe 133m	9.9×10^{-5}
Br 83	7.0×10^{-4}	Xe 135	2.2
Br 84	7.0×10^{-3}	Xe 135m	4.7
Br 84m	3.0×10^{-4}	Xe 137	9.0×10^{2}
Br 85	1.4×10^{-2}	Xe 138	7.0×10^{2}
Br 86	3.0×10^{-4}	Xe 139	1.5
Br 87	4.2×10^{-4}	Xe 140	2.2×10^{-7}
Kr 83m	3.4×10^{-1}	Th 231*	2.0×10^{-13}
Kr 85	3.2×10^{-5}	Th 234*	2.2×10^{-14}
Kr 85m	1.3×10^{1}	Pa 234m*	1.8×10^{-14}
Kr 87	9.4×10^{1}	U 233*	1.7×10^{-20}
Kr 88	6.3×10^{1}	U 234*	3.4×10^{-18}
Kr 89	5.2×10^{2}	U 235*	4.5×10^{-11}
Kr 90	9.2×10^{-2}	U 236*	1.3×10^{-16}

TABLE 4-34

(continued)

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
I 128	2.8×10^{-7}	U 237*	7.1×10^{-10}
I 130	2.0×10^{-7}	U 238*	1.1×10^{-10}
I 131	1.1×10^{-5}	U 239*	6.1×10^{-5}
I 132	3.2×10^{-4}	Np 237*	4.4×10^{-21}
I 133	3.0×10^{-4}	Np 239*	1.5×10^{-7}
I 134	1.0×10^{-2}	Np 240*	3.5×10^{-16}
I 135	4.6×10^{-3}	Pu 239*	4.1×10^{-17}

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* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

activities generated in the criticality are assumed released to the building, and these are attenuated by a factor of 10^{-5} as they pass through two HEPA filters in series during normal operation. In the event of simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the cell is assumed to be released to the atmosphere. The neutron and gamma "shine" from the event is neglected as a source of dose to the population, since these radiations are attenuated by at least five ft. of concrete before emerging from the facility.

Likelihood

The likelihood of a criticality event in a generic fuel reprocessing plant resulting in 10^{18} fissions has been estimated to be approximately 3 x 10^{-5}/year.[9]

However, an analysis of the incidents on record in fuel fabrication facilities results in a probability for accidental criticality of ~8 x 10^{-3}/plant-year (four solution criticality incidents in 490 plant-years of operation).[8] Although an analogy can be drawn between scrap recovery operations in fabrication and spent fuel reprocessing, it should be noted that there have been no criticality incidents on record since 1968.

The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term - Radionuclides given in Table 4-34 to air for normal filter operation and in the event of filter failure; Likelihood ≈ 8 x 10^{-3} to 3 x 10^{-5}/plant-year for normal filter operation, 8 x 10^{-6} to 3 x 10^{-8} for simultaneous failure of two series HEPA filters.

4.7 Mixed Oxide Fuel Fabrication

The accidents considered in mixed oxide fuel fabrication, keyed to the accident categories given in Table 4-1, are listed in Table 4-35.

The risk associated with other incidents, some of which have occurred in the past, is judged to be insignificant in comparison with the accidents considered in Table 4-35. These include sintering furnace explosions, metallographic glove box explosions, autoclave explosions, and ventilation problems from, for example, loss of electrical power. These have resulted in excessive airborne concentrations in work areas, but the release to the environment is generally inconsequential. Filter failures in the ventilation streams have occurred from time to time,[32] but the filters are generally replaced before the time integrated release becomes a significant fraction of the annual release from normal operations.

Current regulatory criteria for the design of new mixed oxide fabrication facilities require that the structures, systems, and other components withstand the effects of natural phenomena. These include all floods, tornados, earthquakes, or missiles of intensity more severe than experienced historically in the location of the plant. Although events with intensities outside of this range are conceivable, the exceedingly low probabilities associated with their occurrence were not evaluated in the current study.

Nine commercial facilities are currently licensed for the production of fuels containing plutonium. However, since the operations employed in mixed oxide fuel fabrication are similar to those utilized in oxide fuel

TABLE 4-35

MIXED OXIDE FUEL FABRICATION ACCIDENTS

A.1	Explosion in oxidation-reduction scrap furnace
B.1	Major facility fire
B.2	Fire in waste compaction glove box
B.3	Ion exchange resin fire
B.4	Dissolver fire in scrap recovery
C.1	Glove failure
C.2	Severe glove box damage
D.1	Criticality

fabrication in general, the fabrication facilities listed in Section 4.5 are deemed germane to the consideration of accident likelihoods in mixed oxide fuel fabrication. A total of 490 plant-years of fuel fabrication experience is derived from Table 4-20.

Most of the data presented in this section were developed in an earlier assessment of effluents from a generic mixed oxide fabrication facility.[8]

A.1 Explosion in Oxidation-Reduction Scrap Furnace

Scrap mixed oxide pellets are conditioned for recycle by heating successively in air and a reducing atmosphere containing hydrogen.* The hydrogen may be mixed with nitrogen or another inert gas, and the hydrogen concentration is controlled to prevent the buildup of an explosive mixture. However, a failure of controls on the mixture could allow pure hydrogen to reach the hot, airfilled furnace after the oxidation step. An explosion in the furnace might damage the glove box and release mixed oxide pellets and dust into the room.

Source Term

It is assumed that approximately 150 kg of pellets and 1.5 kg of dust are released to the room.[42] However, the airborne concentration of heavy particles in the respirable range appears to be limited to approximately 100 mg/m^3.[8] For a room volume assumed to be approximately 5000 m^3, the quantity of mixed oxide released to the building ventilation system would be approximately 500 gm.

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* A reducing atmosphere is also used during the sintering of pellets. However, by virtue of the integral form of the pellets at this stage of the process, the consequences of a postulated explosion in the sintering furnace is judged to be insignificant in comparison with a similar event in the reduction furnace.

The building ventilation system in a generic mixed oxide fabrication plant is assumed to incorporate two HEPA filters in series providing, under normal operations, an attenuation of the source to the environment of approximately 10^{-5}.[8] In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the atmosphere. The mixed oxide is assumed to contain 4.4% plutonium oxide and 95.6% natural uranium oxide. The estimated activities of radionuclides released to the environment are given in column 4, Table 4-36, based upon the specific activities of the components of the mixed oxide in column 3.[5] Uranium isotopes are neglected since they contribute negligibly to dose.

Likelihood

An estimate of $\lesssim 5 \times 10^{-2}$/plant-year has been made for the likelihood of a hydrogen explosion in a fuel fabrication sintering furnace.[8] The same study derived an estimate of $\sim 10^{-3}$/plant-year for chemical explosions in general.

At least one accident has occurred (at a sintering furnace) resulting from detonation of an explosive mixture of hydrogen and oxygen.[8,31] On the basis of the ~490 plant-years of fuel fabrication experience, this would result in a likelihood of $\gtrsim 2 \times 10^{-3}$/plant-year.

TABLE 4-36

RADIONUCLIDE RELEASE RESULTING FROM AN EXPLOSION IN THE OXIDATION-REDUCTION SCRAP FURNACE AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Wt. %	Specific Activity (Ci/gm MO)	Activity Released (Ci)*
Pu 236	3.08×10^{-4}	1.66×10^{-3}	8.3×10^{-6}
Pu 238	8.36×10^{-2}	1.43×10^{-2}	7.2×10^{-5}
Pu 239	2.38	1.48×10^{-3}	7.4×10^{-6}
Pu 240	1.10	2.50×10^{-3}	1.3×10^{-5}
Pu 241	5.28×10^{-1}	5.99×10^{-1}	3.0×10^{-3}
Pu 242	3.08×10^{-1}	1.32×10^{-5}	6.6×10^{-8}

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* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

In another accident evaluation,[45] the probability of a hydrogen explosion in the sintering furnace has been crudely estimated to be lower than 10^0 to 10^{-1}/year and higher than 10^{-1} to 10^{-3}/year.

The likelihood of a simultaneous failure of the two series HEPA filters[9] has been further estimated to be roughly 10^{-3}/demand.

Estimate: Source term - Radionuclides given in Table 4-36 to air for normal filter operation and in the event of filter failure; Likelihood $\gtrsim 5 \times 10^{-2}$ *to* 2×10^{-3}*/plant-year for normal filter operation,* 5×10^{-5} *to* 2×10^{-6}*/plant-year for simultaneous failure of two series HEPA filters.*

B.1 Major Facility Fire

Since the combustible inventory at a mixed oxide fabrication plant is limited, the chance of a catastrophic fire is remote. Also, the use of glass windows in the glove boxes and of fire-suppression equipment both on the inside and outside of the glove boxes is expected.[42] The glove box lines shall also have suitable barriers to isolate various portions of the lines. Nonetheless, a general facility fire is assumed to be initiated by the ignition of a solvent container which is improperly handled. It is assumed that the fire spreads to a large fraction of the glove boxes in several parallel fabrication lines.

Source Term

The bulk of the plutonium in the facility would be in storage in a fire-proof vault. It is estimated that approximately five times the daily output, about 5000 kg mixed oxide would be in process. It is estimated that approximately one-half of the inprocess inventory would be in the dispersible form, and 1% of the dispersible material would be released to the ventilation system during a major facility fire.

The final barrier, composed of two HEPA filters in series and providing an attenuation in source strength of approximately 10^{-5}, would remain intact.[8] In the event of a simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the environment. From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated qualities of radionuclides released to the environment are given in Table 4-37.

Likelihood

The likelihood of a major facility fire in a generic mixed oxide fuel fabrication plant has been estimated to be approximately 2×10^{-4}/year.[8] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term - Radionuclides given in Table 4-37 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 2 \times 10^{-4}$*/plant-year for normal filter operation,* 2×10^{-7}*/plant-year for simultaneous failure of two series HEPA filters.*

TABLE 4-37

RADIONUCLIDE RELEASE RESULTING FROM A MAJOR FIRE AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	4.2×10^{-4}
Pu 238	3.6×10^{-3}
Pu 239	3.7×10^{-4}
Pu 240	6.3×10^{-4}
Pu 241	1.5×10^{-1}
Pu 242	3.3×10^{-6}

---

* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

B.2 Fire in Waste Compaction Glove Box

Burnable wastes are collected at a compaction station until enough has been collected to fill a 55-gallon drum. Waste is composed of polyethylene bagging material, rubber gloves, and cellulose wipes. A solvent-damp wipe in the waste could ignite due to the discharge of static electricity. The fire could breach the containment of the glove box and spread mixed oxide in the dispersible form to the room.

Source Term

It is assumed that the box is at roughly one-half full capacity at the time of the fire, or roughly 50 lbs. of waste. It is further assumed that the waste is contaminated to the extent of 0.01% plutonium by weight.[42] On this basis, approximately 50 gms of mixed oxide would be released to the room. Taking into account the factor of 10^{-5} attenuation of the two series HEPA filters operating normally in the building ventilation system, approximately 5×10^{-4} gms of mixed oxide would be released to the environment. In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the environment. From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated quantities of radionuclides released to the environment are given in Table 4-38.

Likelihood

The likelihood of a local fire in a generic mixed oxide fuel fabrication plant has been estimated to be approximately 10^{-2}/year.[8] Moreover, from

TABLE 4-38

RADIONUCLIDE RELEASE RESULTING FROM A FIRE IN THE WASTE COMPACTION GLOVE BOX AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	8.3×10^{-7}
Pu 238	7.2×10^{-6}
Pu 239	7.4×10^{-7}
Pu 240	1.3×10^{-6}
Pu 241	3.0×10^{-4}
Pu 242	6.6×10^{-9}

---

* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

the incidents on record,[32] there have been at least five fires in fuel fabrication facilities. On the basis of 490 plant-years of fuel fabrication activities, this results in a probability of about 10^{-2}/year. The likelihood of simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

__Estimate:__ Source term - Radionuclides given in Table 4-38 to air for normal filter operation and in the event of filter failure; Likelihood ≈ 10^{-2}/plant-year for normal filter operation, 10^{-5}/year for simultaneous failure of two series HEPA filters.

B.3 Ion Exchange Resin Fire

Anion exchange may be used in scrap recovery to recover chemically contaminated plutonium. Thermal.transients may occur in the resin from radiolytic heating, excessive external heating, or a resin-nitrate reaction. A fire is prevented by process and administrative controls which limits the acidity and temperature of the resin to less than 135°C. However, should these controls fail, and an excursion go unchecked, the ion exchange column could become pressurized and rupture to discharge the resin and contained solution.

Source Term

It is assumed that the column inventory at the time of the rupture is 1400 gms of plutonium,[8] equivalent to approximately 32 kg. of mixed oxide. The fire is assumed to render 0.5 to 0.7% of the mixed oxide airborne, approximately 80% of which is in the respirable range.[8] This results in approximately 150 gms of mixed oxide made airborne within the facility, or 1.5 x 10^{-3} gms released to the environment, taking into account the factor of 10^{-5} attenuation of the two series HEPA filters operating normally in the building ventilation system.[8] In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the atmosphere. From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated quantities of radionuclides released to the environment are given in Table 4-39.

Likelihood

The likelihood of an ion exchange resin fire in a generic mixed oxide fuel fabrication plant has been estimated to be $\lesssim 10^{-1}$/year.[8] However, the likelihood of an ion-exchange resin fire in a generic fuel reprocessing facility has been estimated to be approximately 10^{-4}/year.[9]

Four incidents have been reported, resulting in a release of radioactivity as a result of thermochemical instabilities in ion-exchange processing.[8] On the basis of 490 plant-years of fuel fabrication activities, this history would result in a likelihood of $\sim 8 \times 10^{-3}$/plant-year.

The likelihood of simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

TABLE 4-39

RADIONUCLIDE RELEASE RESULTING FROM A FIRE IN AN ION EXCHANGE RESIN COLUMN AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	2.5×10^{-6}
Pu 238	2.1×10^{-5}
Pu 239	2.2×10^{-6}
Pu 240	3.8×10^{-6}
Pu 241	9.0×10^{-4}
Pu 242	2.0×10^{-8}

---

* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

__Estimate:__ Source term - Radionuclides given in Table 4-39 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 10^{-1}$ *to* 10^{-4}*/plant-year for normal filter operation,* 10^{-4} *to* 10^{-7}*/plant-year for simultaneous failure of two series HEPA filters.*

B.4 Dissolver Fire in Scrap Recovery

After calcining the combustible materials, the scrap and waste from the process are dissolved in nitric acid. Fire around the dissolution tank could heat the liquid and cause it to boil. If the dissolution vessel is closed, it could be ruptured by the internal pressure, and the nitrate would spill on the floor and be dried by the fire.

Source Term

The scrap recovery operation might incorporate an inventory of 25 kg of plutonium,[8] or roughly 500 kg of mixed oxide. The fractional release of mixed oxide into the ventilation system might range from 0.2% to 0.7%, depending upon the duration of the fire and the location of the liquid during the fire.[8] Taking the midrange of these estimates, it is assumed that the fire burns the solution dry, and releases approximately 2.5 kg of mixed oxide to the room air. The source to the environment would be roughly 2.5×10^{-2} gms of mixed oxide, taking into account the 10^{-5} attenuation afforded by the two exit HEPA filters in series operating normally. In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the atmosphere. From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated quantities of radionuclides released to the environment are given in Table 4-40.

Likelihood

The likelihood of a local fire in a generic mixed oxide fuel fabrication plant has been estimated to be ≈ 10^{-2}/year.[8] At least three fires in scrap recovery (not involving ion exchange resin columns) have been recorded in fuel fabrication operations.[32] On the basis of 490 plant-years of fuel fabrication activities, this history would result in a likelihood of approximately 6 x 10^{-3}/plant-year. The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

__Estimate__: Source term - Radionuclides given in Table 4-40 to air for normal filter operation and in the event of filter failure; likelihood ≈ 10^{-2}/plant-year for normal filter operation, 10^{-5}/plant-year for simultaneous failure of two series HEPA filters.

C.1 Glove Failure

Glove failures of various types are a frequent occurrence. They generally involve pinholes or tears, which may result in general contamination of work areas, but a negligible environmental release. Occasionally, a more significant failure, such as an operator inadvertently pulling a glove off of the box, occurs. The glove box might contain the mill used to condition the mixed oxide powder, which is a dusty operation.

TABLE 4-40

RADIONUCLIDE RELEASE RESULTING FROM A DISSOLVER FIRE IN SCRAP RECOVERY AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	4.2×10^{-5}
Pu 238	3.6×10^{-4}
Pu 239	3.7×10^{-5}
Pu 240	6.3×10^{-5}
Pu 241	1.5×10^{-2}
Pu 242	3.3×10^{-7}

---

* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

Source Term

It is assumed that the glove which is involved in the incident contains a gram of mixed oxide powder on the surface, and that approximately 20% of this powder becomes suspended in the room atmosphere.[42] The source to the environment would be approximately 2×10^{-6} gms taking into account the 10^{-5} attenuation of the two series HEPA filters operating normally.[8] In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the atmosphere. From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated quantities of radionuclides released to the environment are given in Table 4-41.

Likelihood

A large glove failure such as described here might be expected to occur once a year.[8] Small failures, such as pin-hole leaks, etc., might be expected to occur more frequently. The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: *Source term - Radionuclides given in Table 4-41 to air for normal filter operation and in the event of filter failure; Likelihood ≈ 1/plant-year for normal filter operation,* 10^{-3}*/plant-year for simultaneous failure of two series HEPA filters.*

TABLE 4-41

RADIONUCLIDE RELEASE RESULTING FROM A GLOVE FAILURE AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	3.3×10^{-9}
Pu 238	2.9×10^{-8}
Pu 239	3.0×10^{-9}
Pu 240	5.0×10^{-9}
Pu 241	1.2×10^{-6}
Pu 242	2.6×10^{-11}

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* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

C.2 Severe Glove Box Damage

Severe mechanical damage to a glove box could rupture the glove box and if the damage occurs in the powder line, mixed oxide powder could be released to the room air. Such an accident might be initiated by a falling beam or crane, or a runaway fork lift truck. Should the incident also breach a compressed air line, the jet of air could provide an additional dispersal mechanism.[8] However, such a sequence of events is not likely to constitute a nominal glove box rupture accident.

Source Term

It is assumed that the glove box involved contains mixed oxide powder, with a batch limit of 11.3 kg of plutonium,[8] or roughly 250 kg of mixed oxide powder. It is further assumed that the initiating event which breaches the glove box contains sufficient energy to disperse a small fraction of the powder. In any case, the airborne concentration of heavy particles in the respirable range would be limited to approximately 100 mg/m^3.[8] For a room volume assumed to be roughly 10^4 m^3,the quantity of mixed oxide released to the building ventilation system would be approximately 1000 gm.

The building ventilation system in the generic plant is assumed to incorporate two HEPA filters in series, providing an attenuation of approximately 10^{-5} in normal operation,[8] resulting in the release to the environment of 1.0×10^{-2} gms of mixed oxide. In the event of simultaneous failure of the two HEPA filters, all of the activity discharged to the room is assumed to be released to the atmosphere.

From the specific activities of mixed oxide radionuclides given in Table 4-36, the estimated quantities of radionuclides released to the environment are given in Table 4-42.

Likelihood

The likelihood of severe glove box damage in a generic mixed oxide fuel fabrication plant has been estimated to be roughly $\gtrsim 10^{-2}$/year.[8] The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term - Radionuclides given in Table 4-42 to air for normal filter operation and in the event of filter failure; Likelihood $\approx 10^{-2}$*/plant-year for normal filter operation,* 10^{-5}*/plant-year for simultaneous failure of two series HEPA filters.*

D.1 Criticality

Criticality could occur at several locations within the mixed oxide plant. Criticality in the dry operations is most likely to occur prior to blending, where undiluted plutonium oxide powder is handled. Scrap recovery, however, is a more probable location for a criticality incident, since the fissile material is moderated by water and transferred as a solution. These high risk operations, however, are recognized in the design and operation of the plant, and equipment is sized to be maintained in a safe geometry or administrative limits are placed on batch sizes.

TABLE 4-42

RADIONUCLIDE RELEASE RESULTING FROM SEVERE GLOVE BOX DAMAGE AT THE MIXED OXIDE FABRICATION PLANT

Nuclide	Activity Released (Ci)*
Pu 236	1.7×10^{-5}
Pu 238	1.4×10^{-4}
Pu 239	1.5×10^{-5}
Pu 240	2.5×10^{-5}
Pu 241	6.0×10^{-3}
Pu 242	1.3×10^{-7}

---

* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

Source Term

Criticality is assumed to occur in the dissolution step of scrap recovery, which may contain as much as 2.5 kg of plutonium.[8] For the ten solution criticalities on record,[31] the average number of fissions on record was $\sim 4 \times 10^{18}$. However, one of the incidents involved a record $\sim 4 \times 10^{19}$ fissions at the Chemical Processing Plant of the Idaho Reactor Testing Area, considered unrepresentative of the fuel fabrication operations. Neglecting this incident, the average number of fissions is $\sim 4 \times 10^{17}$. Other evaluations have selected $\sim 10^{18}$ fissions as representative for criticality in a fuel fabrication plant,[8,16] and this estimate will be adopted here as well.

The radionuclides released ten minutes after an incident involving 10^{18} fissions, based upon ORIGEN calculations (for a ^{235}U criticality incident)[44] are given in Table 4-43. These estimates further assume 100% of the noble gases and 50% of the halogens released to the environment. Approximately 0.2% of the activities created in the excursion are released to the building air. Additionally, some release of plutonium in the solution can be expected. For an excursion terminated following the evaporation of 10 liters of excess solution, containing 150 gms of Pu per liter, approximately 0.2%, or 3 gms of plutonium are assumed to be rendered airborne.[8] This contribution to the source term is included in Table 4-43, following an attenuation by a factor of 10^{-5} from two stages of HEPA filtration operating normally in series.[8] Additionally, the neutron and gamma radiation associated with a burst of 10^{18} fissions

TABLE 4-43

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT AT THE MIXED OXIDE FUEL FABRICATION FACILITY

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	1.2×10^{-1}	I 136	2.3×10^{1}
Br 80m	5.7×10^{-5}	I 137	3.3×10^{-4}
Br 82	1.9×10^{-5}	Xe 133	1.7×10^{-3}
Br 82m	3.2×10^{-3}	Xe 133m	9.9×10^{-5}
Br 83	3.5	Xe 135	2.2
Br 84	3.5×10^{1}	Xe 135m	4.7
Br 84m	1.5	Xe 137	9.0×10^{2}
Br 85	7.2×10^{1}	Xe 138	7.0×10^{2}
Br 86	1.5	Xe 139	1.5
Br 87	2.1	Xe 140	2.2×10^{-7}
Kr 83m	3.4×10^{-1}	Th 231*	2.0×10^{-13}
Kr 85	3.2×10^{-5}	Th 234*	2.2×10^{-14}
Kr 85m	1.3×10^{1}	Pa 234m*	1.8×10^{-14}
Kr 87	9.4×10^{1}	U 233*	1.7×10^{-20}
Kr 88	6.3×10^{1}	U 234*	3.4×10^{-18}
Kr 89	5.2×10^{2}	U 235*	4.5×10^{-11}
Kr 90	9.2×10^{-2}	U 236*	1.3×10^{-16}
I 128	1.4×10^{-3}	U 237*	7.1×10^{-10}

TABLE 4-43

(continued)

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
I 130	1.0×10^{-3}	U 238*	1.1×10^{-10}
I 131	5.5×10^{-2}	U 239*	6.1×10^{-5}
I 132	1.6	Np 237*	4.4×10^{-21}
I 133	1.5	Np 239*	1.5×10^{-7}
I 134	5.2×10^{1}	Np 240*	3.5×10^{-16}
I 135	2.3×10^{1}	Pu 236*	1.1×10^{-6}
		Pu 238*	9.7×10^{-6}
		Pu 239*	1.0×10^{-6}
		Pu 240*	1.7×10^{-6}
		Pu 241*	4.1×10^{-4}
		Pu 242*	9.0×10^{-9}

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* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

would result in a dose to the population. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

Likelihood

The likelihood of a criticality incident in a mixed oxide fuel fabrication plant has been estimated to be approximately 8 x 10^{-3}/plant-year.[8]* As discussed in Reference 8, an improved basis for such an estimate would consider the total fuel throughput, the fuel forms during processing and the fuel reactivities involved.

In a separate study for a generic fuel reprocessing facility, the likelihood of criticality has been estimated to be approximately 3 x 10^{-5}/year.[9]

The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/demand.[9]

Estimate: Source term - Radionuclides given in Table 4-43 to air for normal filter operation and in the event of filter failure plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood ≈ 8 x 10^{-3} to 3 x 10^{-5}/plant-year for normal filter operation, 8 x 10^{-6} to 3 x 10^{-8}/plant-year for simultaneous failure of two series HEPA filters.

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[*] The estimate contained in Reference 8 has been revised to reflect the increase in experience to 490 plant-years.

4.8 Plutonium Storage

The only accident considered in the generic plutonium storage facility is criticality. The solid plutonium oxide is neither flammable nor explosive, and no flammable or explosive compounds are expected in significant amounts in the storage area. It is not possible for overheating of the plutonium to occur, since the material is cooled by natural convection air. Spills are unlikely when the material is in the solid state, and if they were to occur, the contamination would be contained largely within the building with negligible risk to the environment. The building structures will be engineered to resist earthquakes and tornadoes, and the storage site will be selected to withstand the maximum credible flood. Natural events outside of the range of design basis tornadoes, earthquakes, or floods are not considered within the scope of the current study.

There are no existing storage facilities for plutonium of the capacity of the generic plutonium storage facility. A few small storage facilities located at or in the vicinity of fuel fabrication or reprocessing plants handle plutonium in the liquid form as a plutonium nitrate solution. Thus there does not exist an adequate data base to assess the likelihood of accidents at the generic plutonium storage facility postulated in this study.

D.1 Criticality

The generic plutonium storage facility is envisioned to store plutonium in carefully analyzed, predetermined arrays. Because of the quiescent, uniform nature of storage, with no other operations planned, the likelihood of criticality is extremely remote. A number of storage containers would have to be forcefully ruptured or crushed simultaneously and their contents would have to be collected in a favorable geometry for criticality to occur.

Source Term

For the 26 criticality incidents on record,[31] the total number of fissions range from ~3 x 10^{15} to 4 x 10^{19}. For the 11 non-solution criticalities, considered more representative of a postulated criticality incident involving solid plutonium storage, the average number of fissions is ~10^{17}. The radionuclides released ten minutes after an incident involving 10^{17} fissions, based upon ORIGEN calculations,[44] are given in Table 4-44. These estimates further assume 100% of the noble gases and 50% of the halogens released to the environment. Approximately 0.2% of the actinides created in the excursion are released to the building air. Additionally, some release of the plutonium powder in the containers can be expected from the excursion. It is assumed that the energy release in the criticality would be sufficient to rupture one of the storage containers and release 1% of the 4.3 kg of plutonium in the container to the building atmosphere. This contribution to the source term is included in Table 4-44, following an attenuation by a factor of 10^{-5} from two stages of HEPA filtration in series during normal operation.[8] In the event of simultaneous failure of the two HEPA filters, all of the particulate activity discharged to the building is assumed to be released to the atmosphere.

TABLE 4-44

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT AT THE PLUTONIUM STORAGE FACILITY

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	1.2×10^{-2}	I 136	2.3
Br 80m	5.7×10^{-6}	I 137	3.3×10^{-5}
Br 82	1.9×10^{-6}	Xe 133	1.7×10^{-4}
Br 82m	3.2×10^{-4}	Xe 133m	9.9×10^{-6}
Br 83	3.5×10^{-1}	Xe 135	2.2×10^{-1}
Br 84	3.5	Xe 135m	4.7×10^{-1}
Br 84m	1.5×10^{-1}	Xe 137	9.0×10^{1}
Br 85	7.2	Xe 138	7.0×10^{1}
Br 86	1.5×10^{-1}	Xe 139	1.5×10^{-1}
Br 87	2.1×10^{-1}	Xe 140	2.2×10^{-8}
Kr 83m	3.4×10^{-2}	Th 231*	2.0×10^{-14}
Kr 85	3.2×10^{-6}	Th 234*	2.2×10^{-15}
Kr 85m	1.3	Pa 234m*	1.8×10^{-15}
Kr 87	9.4	U 233*	1.7×10^{-21}
Kr 88	6.3	U 234*	3.4×10^{-19}
Kr 89	5.2×10^{1}	U 235*	4.5×10^{-12}
Kr 90	9.2×10^{-3}	U 236*	1.3×10^{-17}
I 128	1.4×10^{-4}	U 237*	7.1×10^{-11}
I 130	1.0×10^{-4}	U 238*	1.1×10^{-11}

TABLE 4-44

(continued)

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
I 131	5.5×10^{-3}	U 239*	6.1×10^{-6}
I 132	1.6×10^{-1}	Np 237*	4.4×10^{-22}
I 133	1.5×10^{-1}	Np 239*	1.5×10^{-8}
I 134	5.2	Np 240*	3.5×10^{-17}
I 135	2.3	Pu 236*	1.6×10^{-6}
		Pu 238*	1.4×10^{-4}
		Pu 239*	1.4×10^{-5}
		Pu 240*	2.4×10^{-5}
		Pu 241*	5.9×10^{-3}
		Pu 242*	1.3×10^{-7}

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* The values tabulated are for normal operation of the two HEPA filters. In the event of simultaneous failure of the filters, these values are increased by a factor of 10^5.

Additionally, the neutron and gamma radiation associated with a burst of 10^{17} fissions would result in a dose to the population. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

Likelihood

The likelihood of a criticality incident in a mixed oxide fuel fabrication plant has been estimated to be approximately 8×10^{-3}/plant year[8*]. The extrapolation of a probability estimated for fuel fabrication operations to a storage situation is questionable, but no other data exist. The only criticalities on record in fuel fabrication facilities occurred in solution. Dry criticalities have only occurred in reactor experiments. Since plutonium storage operations are dry, the likelihood of criticality is estimated to be at least an order of magnitude lower than the above estimate. Moreover, fuel fabrication operations are active, whereas plutonium storage is a quiescent operation. This factor is assumed to reduce the likelihood of criticality by at least another order of magnitude. The likelihood of a simultaneous failure of the two series HEPA filters has been further estimated to be roughly 10^{-3}/ demand.[9]

Estimate: *Source term - Radionuclides given in Table 4-44 to air for normal filter operation and in the event of filter failure plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood* $\lessapprox 8 \times 10^{-5}$*/plant-year for normal filter operation,* 8×10^{-8}*/plant-year for simultaneous failure of two series HEPA filters.*

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* The estimate contained in Reference 8 has been revised to reflect the increase in experience to 490 plant/years.

4.9 Transportation

The accidents considered in transportation of fuel cycle materials are given in Table 4-45. The accident category format given in Table 4-1 is not considered appropriate to the consideration of transportation accidents.

Only improperly closed plutonium oxide packages have been considered at the low consequence end of the accident spectrum since the risk associated with the release of ore or ore concentrate is insignificant, and the remaining fuel cycle materials are not dispersible in the absence of a driving force (with the exception of irradiated fuel coolant leaks, which are analyzed). Similarly, the risk associated with accidents involving uranium hexafluoride, irradiated fuel, and plutonium oxide are considered to dominate the collision considerations. A release from a spent fuel shipment involved in an accident which submerses the cask in deep water has been neglected because the risk associated with a severe collision is considered to be more significant. Similarly, the likelihood of complete loss of coolant to irradiated fuel is considered to be vanishingly small except under severe collision conditions.

Criticality incidents involving the shipment of fissile material are considered separately, although they are assumed to occur only under extra severe collision conditions. Criticality involving irradiated fuel is not considered credible because of the relatively low fissile material concentration in the fuel.

TABLE 4-45

TRANSPORTATION ACCIDENTS

1. Leakage of coolant from irradiated fuel cask
2. Improperly closed plutonium oxide container
3. Release from a collision involving natural uranium hexafluoride
4. Release from a collision involving enriched uranium hexafluoride
5. Release from a collision involving irradiated fuel
6. Release from a collision involving irradiated fuel followed by release of fuel from the cask
7. Release from a collision involving plutonium oxide
8. Criticality of unirradiated fuel
9. Criticality of enriched UO_2
10. Criticality of plutonium oxide

The reliance for safety in transport of radioactive materials is placed on packaging. The packaging must conform with regulatory standards established by the Department of Transportation, the Nuclear Regulatory Commission, and some of the States. The standards require the packaging to prevent the loss or dispersal of the radioactive contents, retain the shielding efficiency, assure criticality safety, and provide adequate heat dissipation. Type A packaging, which may be used to transport uranium concentrate, natural uranium hexafluoride, and low level wastes, is unlikely to be breached in a minor accident, and some fraction would not be breached in very severe accidents. Type B packaging, used for the remaining materials transported in the fuel cycle, is likely to withstand all but very severe accidents.

In the past 25 years, approximately 2600 packages of irradiated fuel have been transported in routine commerce.[42] The number of shipments of the other materials in the fuel cycle have not been well documented.

Much of the data presented in this section was developed in two earlier studies. The first was a survey of the transportation of radioactive materials to and from nuclear power plants,[10] and the second was an assessment of transportation risks in the nuclear power industry.[48]

1. Leakage of Coolant from Irradiated Fuel Cask

A rail cask carries up to 18 LWR fuel assemblies. The cask must provide the means to dissipate the heat produced by radioactive decay. In the usual configuration, water coolant is sealed in the cask and the heat is dissipated through natural convection to the surface, which is cooled in turn by natural or forced air convection over fins. The rugged, leaktight design of the cask, coupled with rigid procedures imposed on the shipper, prevent leaks from occurring. Also, each cask is held at the origin until checks have been made on pressure, temperature and leakage. However, a small leak which is undetectable by visual observation could go unchecked for some time.

Source Term

A previous analysis of the consequences of a leak in a spent fuel shipping cask indicated that a leakage rate of approximately 0.001 cm^3/sec. is the largest that can go undetected by visual observation.[10] Based upon 0.25% of the fuel rods being perforated, it was estimated that approximately 1 μCi/cm^3 of gross fission product activity might be in the cask coolant, and that this activity is primarily Cs-137. In five days, approximately 400 μCi of activity would be released to the surface of the cask, and approximately 1% of this activity is assumed to be released to the environment as an aerosol.

Likelihood

Of the 3600 shipments of spent fuel, there have been no incidents recorded involving a leak of coolant from the cask. Accordingly, lacking any other data to estimate the probability of a small leak, it is estimated that the likelihood of a small leak of irradiated fuel coolant is $\simeq 3 \times 10^{-4}$/shipment.

Estimate: Source term ≈ 4 μCi ^{137}Cs to air; Likelihood ≈ 3 x 10^{-4}/ shipment.

2. Improperly Closed Plutonium Oxide Container

Current package designs for plutonium oxide incorporate sealed metal cans within an inner, gasketed steel container supported within an outer steel drum of from 10 to 100 gallon capacity. Current package designs are expected to be modified to dissipate up to a maximum of 100 watts of heat, allowing container loadings of up to 6.4 kg Pu.[5] It is envisioned that 40 of these containers would comprise a truck shipment. The generic shipping distance is estimated to be 300 miles from the reprocessing facility to the plutonium storage facility and an equal distance to the mixed oxide fabrication plant. A small amount of dispersible plutonium might be released to the environment from an improperly closed package.

Source Term

An improperly closed package containing plutonium oxide would release only a small amount of its contents because of the absence of a driving force and the several barriers between the powder and the environment. It is assumed that in the event of a complete breach of containment, 0.1% of the plutonium oxide would be released from the package. According to estimates made by Mishima,[49] the fractional airborne release of plutonium dioxide particles less than 10 μm from either a stainless steel or asphalt-gravel aggregate surface at a nominal wind speed of 2.5 mph is approximately 8 x 10^{-4} in 24 hours. Thus for a 6.4 kg container, approximately 5.1 mg of plutonium might be released to the environment. The estimated activities of the plutonium isotopes released are given in Table 4-46.

TABLE 4-46

RADIONUCLIDE RELEASE RESULTING FROM AN IMPROPERLY CLOSED PLUTONIUM OXIDE CONTAINER

Nuclide	Activity Released (Ci)
Pu 236	1.9×10^{-4}
Pu 238	1.7×10^{-3}
Pu 239	1.8×10^{-4}
Pu 240	2.9×10^{-4}
Pu 241	7.0×10^{-2}
Pu 242	1.5×10^{-6}

Likelihood

It has been estimated that for Type B packages, approximately 1 in 100,000 might be improperly closed when shipped.[10] Then, on the basis of 40 containers per shipment, the likelihood of improper closure is 4×10^{-4}/shipment. Also in a recent survey of package closure faults,[50] while several container packaging errors were observed, there was no instance of complete loss of container integrity in 775 shipments.

Estimate: Source term - Radionuclides given in Table 4-46 to air. Likelihood $\approx 1 \times 10^{-3}$*/shipment to* 4×10^{-4}*/shipment.*

3. Release from a Collision Involving Natural Uranium Hexafluouride

Natural UF_6 is typically shipped as a solid from the UF_6 conversion plant to the enrichment facility in cylinders containing 12.7 MT of UF_6. The generic shipping distance is estimated to be 500 miles and only one cylinder would be shipped in a truck. A collision resulting in a fire could volatilize and release the UF_6 to the environment.

Source Term

For a fire of long duration following a collision, the entire contents of the cylinder could be volatilized and released to the environment. This would result in approximately 8600 kg of freshly separated uranium discharged to the atmosphere. The isotopic composition of this uranium is given in Section 4.3.

Likelihood

Although the cylinder could conceivably be breached by a less than severe accident, it would take a severe accident to produce a fire of sufficient duration to vaporize the entire contents of the cylinder. The accident probabilities for truck, rail, and barge accidents of various severities are given in Table 4-47.[10] For a shipping distance of 500 miles, then, the likelihood of an accident resulting in a fire which volatilizes the UF_6 is approximately 4×10^{-6}.

Estimate: Source term $\cong 2.8$ *Ci* ^{238}U, 1.3×10^{-1} *Ci* ^{235}U, 3.0 *Ci* ^{234}U, 2.8 *Ci* ^{234}Th *to air; Likelihood* $\cong 4 \times 10^{-6}$*/shipment.*

4. Release from a Collision Involving Enriched Uranium Hexafluoride

Enriched UF_6 is shipped as a solid from the enrichment plant to the fuel fabrication facility in cylinders containing 2.2 MT of UF_6. The generic shipping distance is estimated to be 750 miles and typically five cylinders would comprise one truck shipment. A collision resulting in a fire could volatilize and release the UF_6 to the environment.

Source Term

For a fire of long duration following a collision, the entire contents of the cylinders could be volatilized and released to the environment. This would result in approximately 7400 kg of enriched uranium discharged to the atmosphere. The isotopic composition of this uranium is given in Table 4-17.

Likelihood

Although the cylinders could conceivably be breached by a less than severe accident, it would take a severe accident to produce a fire of sufficient duration to vaporize the entire contents of the cylinders. The accident

TABLE 4-47

ACCIDENT PROBABILITIES FOR TRUCK, RAIL, AND BARGE ACCIDENTS OF VARIOUS SEVERITIES[10]

Severity	Accident Probability (per vehicle mile)
Minor	2×10^{-6}
Moderate	3×10^{-7}
Severe	8×10^{-9}
Extra Severe	2×10^{-11}
Extreme	1×10^{-13}

probabilities for truck, rail, and barge accidents of various severities are given in Table 4-47.[10] For a shipping distance of 750 miles, then, the likelihood of an accident resulting in a fire which volatilizes the UF_6 is approximately 6×10^{-6}.

Estimate: Source term ≈ *2.4 Ci ^{238}U, 2.8 Ci ^{237}U, 5.3×10^{-1} Ci ^{236}U, 4.1×10^{-1} Ci ^{235}U, 17 Ci ^{234}U to air; Likelihood* ≈ *6×10^{-6}/shipment.*

5. Release from a Collision Involving Irradiated Fuel

Approximately 3.7 MT of irradiated fuel, comprising from 7 to 18 LWR fuel assemblies, are transported in a cask from the reactor to the reprocessing plant after a minimum of 150 days cooling. The spent fuel cask is designed to provide shielding, criticality safety, heat dissipation, and protection against severe accidents. The cask is assumed to be transported for 20 miles by truck to the rail head, where it is shipped 1000 miles to the reprocessing facility by rail. An extra severe collision coupled with a fire could cause a breach in the cask, releasing a fraction of the contained fission products to the environment.

Source Term

The estimated quantities of fission products released to the environment in the event of a collision involving irradiated fuel are given in Table 4-48. The values given are based upon release fraction estimates from References 10 and 48. These sources are in agreement in their predictions of the release fractions for Kr-85 (~3%) and iodine (~0.2%). Their estimates for the release fractions of the remaining fission products, however, differ by

TABLE 4-48

RADIONUCLIDE RELEASE RESULTING FROM AN EXTRA SEVERE COLLISION INVOLVING IRRADIATED FUEL

Nuclide	Activity Released (Ci)
Kr 85	1200
I 129	1.5×10^{-5}
I 131	1.6×10^{-2}
Ru 103	27
Ru 106	140
Zr 95	80
Nb 95	150
Sr 89	20
Sr 90	19
Y 90	19
Y 91	42
Cs 134	55
Cs 137	27
Ce 141	18
Ce 144	200
Pm 147	32

an order of magnitude. Reference 10 assumes that $\sim 10^{-3}$% of the remaining fission products are released to the environment, whereas Reference 48 adopts the more conservative value of 10^{-2}%. Accordingly, the values in Table 4-48 assume a release fraction of 5×10^{-5}. Moreover, the isotopic breakdown of fission products other than krypton and iodine given in Table 4-48 is based upon the tabulated activities in the fuel given in Table 4-24.

Likelihood

According to Reference 10, an "extra severe" accident would be required for a break to occur in a spent fuel shipping cask. The probability of an extra severe accident, as tabulated in Table 4-47, is 2×10^{-11}/vehicle-mile. However, according to a more recent analysis involving fault tree analysis,[48] the average accident involving a typical release of radioactivity is expected to have a probability of 4.6×10^{-9}/mile for a truck shipment, and 9.6×10^{-9}/mile for a rail shipment. The typical shipment of spent fuel travels a distance of 20 miles by truck and 1000 miles by rail.

Estimate: Source term - Radionuclides given in Table 4-48 to air; Likelihood $\approx 9 \times 10^{-6}$*/shipment to* 2×10^{-8}*/shipment.*

6. Release from a Collision Involving Irradiated Fuel Followed by Release of Fuel from the Cask

It is possible, although highly improbable, in the event of a collision, that the irradiated fuel cask could be damaged to the extent that one or more of the fuel elements would be released from the cask.

Source Term

The release magnitude of radionuclides to the environment is assumed to be similar to that given in Table 4-48. However, if irradiated fuel elements are released from the cask, a significant dose to the population results from the direct radiation "shine" emitted by the decaying fission products in the fuel. Assuming that seven irradiated fuel assemblies are released from the cask in an extremely severe collision, the radiation level at 100 feet has been estimated to be 10^4 r/hr.[10] This dose is attenuated by the inverse square relationship and a dry air removal cross section, as described in Section 3.2, to arrive at a population dose from direct shine. It is further assumed that the fuel elements remain unshielded for 10 hours,[10] and that the uniform population density beyond 10 meters is 290 people/mi^2, as given in Table 3-1. It is unrealistic to expect that the population close to the accident would remain exposed to the radiation for the full 10 hours. Accordingly, it is assumed that people within 100 meters of the fuel would be evacuated within 1/2 hour and those within one mile would be evacuated within 2 hours after the accident (a total of $\sim$800 people). The population dose from direct shine is added to the total body dose accruing from exposure to the radionuclides given in Table 4-48.

Likelihood

According to Reference 10, the release of irradiated fuel elements from the cask could only occur under extremely severe accident conditions. Referring to Table 4-47, the likelihood of an extremely severe accident is roughly a factor of 10^{-2} less likely than that of an accident of sufficient severity to breach the shipping container. Applying this factor to the breach likelihoods given in the previous section, the likelihood of a release of fuel elements from the cask for a generic shipment of approximately 1000 miles is estimated to range from 9 x 10^{-8}/ shipment to 2 x 10^{-10}/shipment.

Estimate: Source term - Radionuclides given in Table 4-48 to air plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood ≈ 9 x 10^{-8}/shipment to 2 x 10^{-10}/shipment.

7. Release from a Collision Involving Plutonium Oxide

In the event of a very severe collision, a container holding 6.4 kg of dispersible plutonium oxide might be breached, releasing a fraction of its contents to the environment. The generic shipping distance is estimated to be 300 miles from the reprocessing facility to the plutonium storage facility and an equal distance to the mixed oxide fabrication plant.

Source Term

The estimated quantities of plutonium isotopes released to the environment in the event of a severe accident are given in Table 4-49. These estimates are based upon 0.1% of the material in a container holding 6.4 kg of plutonium oxide being rendered immediately airborne and released to the environment when a container fails in an accident. A fractional release to the environment of 0.1% has been estimated as a nominal value in Reference 51, and as a large release in Reference 48.

TABLE 4-49

RADIONUCLIDE RELEASE RESULTING FROM A COLLISION INVOLVING PLUTONIUM OXIDE

Nuclide	Activity Released (Ci)
Pu 236	2.4×10^{-1}
Pu 238	2.1
Pu 239	2.3×10^{-1}
Pu 240	3.6×10^{-1}
Pu 241	8.8×10^{1}
Pu 242	1.9×10^{-3}

Likelihood

According to Reference 42, even under extra severe collision conditions, plutonium oxide containment is not considered to be breached. However, two recent studies have derived, using fault tree analysis, finite probabilities of release in the event of an accident. Reference 51 estimates one accident involving a release for every 220 accidents during transport. Coupling this estimate with the estimated truck accident probability of 2.5×10^{-6}/mile, a release likelihood of 1.1×10^{-8}/mile is derived. A considerably lower estimate of approximately 7×10^{-12}/mile is obtained in Reference 48 corresponding to the release fraction of 0.1%. Using an average shipment distance of 300 miles, the likelihood of a release, then, lies in the range of 3×10^{-6}/shipment to 2×10^{-9}/shipment.

Estimate: *Source term - Radionuclides given in Table 4-49 to air. Likelihood* $\approx 3 \times 10^{-6}$*/shipment to* 2×10^{-9}*/shipment.*

8. Criticality of Unirradiated Fuel

Unirradiated fuel, both fuel assemblies and mixed oxide fuel pins, are shipped by truck in metal containers which support the fuel along its entire length during transport. A single shipment might consist of from six to sixteen packages, each containing the equivalent of two LWR assemblies. The shipping containers are designed for criticality safety under all credible accident conditions, including submersion in water.

Mixed oxide fuel pins are assumed to be transported approximately 200 miles by truck to the fuel fabrication plant and completed assemblies approximately 1000 miles by truck to the reactor. Only under extra severe accident conditions, which could compromise the safe geometry, coupled with the presence of water moderation, would a criticality incident be possible.

Source Term

For the 26 criticality incidents on record,[31] the total number of fissions range from $\sim 3 \times 10^{15}$ to 4×10^{19}. For the 11 non-solution incidents, considered more representative of a postulated criticality involving unirradiated fuel in transport, the average number of fissions is $\sim 10^{17}$. However, criticality in unirradiated fuel is not expected to cause any release of radioactive materials from the fuel pins, since the fuel cladding would retain the fission products created in the excursion.[10] The only anticipated source of exposure to the general population beyond 10 meters from the excursion would be from the prompt neutron and gamma radiation associated with the burst. The methodology for evaluating the population dose under these conditions is discussed in Section 3.2.*

Likelihood

A criticality has never occurred in the transportation of fissile materials. For a criticality to occur, a rearrangement of the fuel into a favorable geometry must take place. It is assumed that such an event could only take place under extremely severe accident conditions. Referring to Table 4-47, the likelihood of an extremely severe accident is roughly a factor of 10^{-2} less likely than that of an accident of sufficient severity to breach the shipping container. Lacking estimates for a fresh fuel container, it is assumed that the likelihood of an accident of sufficient severity to breach the container is comparable to previous estimates for spent fuel containers, namely 4.6×10^{-9}/mile to 2×10^{-11}/mile for shipments by truck. Thus for the assumed 1000 mile distance from the fabrication facility to the reactor,* the likelihood of a criticality incident for fresh fuel is in the range of 5×10^{-8}/shipment to 2×10^{-10}/shipment.

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* The shielding factor associated with this methodology is retained, since the water necessary for moderation/reflection is assumed to provide shielding equivalent to 8 inches of concrete.

Estimate: Source term - Ionizing radiations associated with a criticality of 10^{17} fissions (see Section 3.2 for dose methodology); Likelihood ≈ 5×10^{-8}/shipment to 2×10^{-10}/shipment.

9. Criticality of Enriched UO_2

Enriched UO_2 powder is shipped by truck in containers designed to prevent criticality under all credible normal transport conditions as well as severe accidents. A generic shipment of uranium dioxide contains approximately 4.5 MT of powder for a distance of 750 miles. Only under extra severe accident conditions, which could compromise the safe geometry, coupled with the presence of water moderation, would a criticality incident be possible.

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* The shipments of mixed oxide fuel pins from the mixed oxide plant to the uranium fabrication plant has the effect of increasing the total travel miles by roughly 7%.

Source Term

Approximately 10^{17} fissions, representative of non-solution criticality incidents, are considered appropriate for a postulated criticality involving enriched UO_2 in transport. The radionuclides released ten minutes after an incident involving 10^{17} fissions, based upon ORIGEN calculations,[44] are given in Table 4-50. These estimates assume that 100% of the noble gases and halogens, and 0.2% of the actinides created in the excursion are released to the environment. Additionally, the neutron and gamma radiation associated with a burst of 10^{17} fissions would result in a dose to the population assumed to be uniformly distributed beyond 10 meters from the excursion. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

Likelihood

A criticality has never occurred in the transportation of fissile materials. For a criticality to occur, the containers of UO_2 would have to depart from the safe geometry provided by the shipping containers. It is assumed that such an event could only occur under extremely severe accident conditions. Referring to Table 4-47, the likelihood of an extremely severe accident is roughly a factor of 10^{-2} less likely than an accident of sufficient severity to breach the shipping container. Lacking estimates for a UO_2 container, it is assumed that the likelihood of an accident of sufficient severity to breach the container is comparable

TABLE 4-50

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT INVOLVING ENRICHED UO_2 IN TRANSPORT

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	2.4×10^{-2}	I 136	4.6
Br 80m	1.1×10^{-5}	I 137	6.6×10^{-5}
Br 82	3.8×10^{-6}	Xe 133	1.7×10^{-4}
Br 82m	6.4×10^{-4}	Xe 133m	9.9×10^{-6}
Br 83	7.0×10^{-1}	Xe 135	2.2×10^{-1}
Br 84	7.0	Xe 135m	4.7×10^{-1}
Br 84m	3.0×10^{-1}	Xe 137	9.0×10^{1}
Br 85	1.4×10^{1}	Xe 138	7.0×10^{1}
Br 86	3.0×10^{-1}	Xe 139	1.5×10^{-1}
Br 87	4.2×10^{-1}	Xe 140	2.2×10^{-8}
Kr 83m	3.4×10^{-2}	Th 231	2.0×10^{-9}
Kr 85	3.2×10^{-6}	Th 234	2.2×10^{-10}
Kr 85m	1.3	Pa 234m	1.8×10^{-10}
Kr 87	9.4	U 233	1.7×10^{-16}
Kr 88	6.3	U 234	3.4×10^{-14}
Kr 89	5.2×10^{1}	U 235	4.5×10^{-7}
Kr 90	9.2×10^{-3}	U 236	1.3×10^{-12}
I 128	2.8×10^{-4}	U 237	7.1×10^{-6}
I 130	2.0×10^{-4}	U 238	1.1×10^{-6}
I 131	1.1×10^{-2}	U 239	6.1×10^{-1}
I 132	3.2×10^{-1}	Np 237	4.4×10^{-16}
I 133	3.0×10^{-1}	Np 239	1.5×10^{-3}
I 134	1.0×10^{1}	Np 240	3.5×10^{-11}
I 135	4.6	Pu 239	4.1×10^{-13}

to previous estimates for a PuO_2 container, namely 1.1×10^{-8}/mile to 7×10^{-12}/mile. Thus for the assumed 750 mile distance from the UO_2 plant to the fabrication facility, the likelihood of a criticality incident for UO_2 is in the range of 8×10^{-8}/ shipment to 5×10^{-11}/shipment.

Estimate: *Source term - Radionuclides given in Table 4-50 to air plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood* $\sim 8 \times 10^{-8}$*/shipment to* 5×10^{-11}*/shipment.*

10. Criticality of PuO_2

Plutonium dioxide is shipped by truck within metal cans contained in steel containers supported within an outer steel drum. Approximately 40 containers, each containing up to 6.4 kg Pu, are contained in a generic shipment of 300 miles from the reprocessing plant to the mixed oxide fabrication plant. The shipping containers are designed to prevent criticality under all conceivable conditions. Only under extra severe accident conditions could the safe geometry be compromised, leading to a potential criticality incident.

Source Term

Approximately 10^{17} fissions, representative of non-solution criticality incidents, is considered appropriate for a postulated criticality involving PuO_2 in transport. The radionuclides released ten minutes after an incident involving 10^{17} fissions, based upon ORIGEN calculations (for a ^{235}U criticality),[44] are given in Table 4-51. These estimates assume that 100% of the noble gases and halogens, and 0.2% of the actinides created in the excursion are released to the environment. Additionally, it is assumed that the energy release would be sufficient to expel approximately 1% of the plutonium in one of the containers. This contribution to the source term is included in Table 4-51. Finally, the neutron and gamma radiation associated with a burst of 10^{17} fissions would result in a dose to the population. The methodology for evaluating this direct "shine" dose is discussed in Section 3.2.

Likelihood

A criticality has never occurred in the transportation of fissile materials. For a criticality to occur, the containers of PuO_2 would have to depart from the safe geometry provided by the shipping containers. It is assumed that such an event would only occur under extremely severe accident conditions. Referring to Table 4-47, the likelihood of an extremely severe accident is roughly a factor of 10^{-2} less likely than an accident of sufficient severity to breach the shipping container. According to estimates given earlier, the likelihood of an accident of sufficient severity to breach the container is in the range of 1.1×10^{-8}/mile to 7×10^{-12}/mile. Thus for the assumed 300 mile distance of the shipment, the likelihood of a criticality incident for PuO_2 is in the range of 3×10^{-8}/shipment to 2×10^{-11}/shipment.

Estimate: Source term - Radionuclides given in Table 4-51 to air plus ionizing radiations (see Section 3.2 for dose methodology); Likelihood $\approx 3 \times 10^{-8}$*/shipment to* 2×10^{-11}*/shipment.*

TABLE 4-51

RADIONUCLIDE RELEASE RESULTING FROM A CRITICALITY INCIDENT INVOLVING PuO_2 IN TRANSPORT

Nuclide	Activity Released (Ci)	Nuclide	Activity Released (Ci)
Br 80	2.4×10^{-2}	I 136	4.6
Br 80m	1.1×10^{-5}	I 137	6.6×10^{-5}
Br 82	3.8×10^{-6}	Xe 133	1.7×10^{-4}
Br 82m	6.4×10^{-4}	Xe 133m	9.9×10^{-6}
Br 83	7.0×10^{-1}	Xe 135	2.2×10^{-1}
Br 84	7.0	Xe 135m	4.7×10^{-1}
Br 84m	3.0×10^{-1}	Xe 137	9.0×10^{1}
Br 85	1.4×10^{1}	Xe 138	7.0×10^{1}
Br 86	3.0×10^{-1}	Xe 139	1.5×10^{-1}
Br 87	4.2×10^{-1}	Xe 140	2.2×10^{-8}
Kr 83m	3.4×10^{-2}	Th 231	2.0×10^{-9}
Kr 85	3.2×10^{-6}	Th 234	2.2×10^{-10}
Kr 85m	1.3	Pa 234m	1.8×10^{-10}
Kr 87	9.4	U 233	1.7×10^{-16}
Kr 88	6.3	U 234	3.4×10^{-14}
Kr 89	5.2×10^{1}	U 235	4.5×10^{-7}
Kr 90	9.2×10^{-3}	U 236	1.3×10^{-12}
I 128	2.8×10^{-4}	U 237	7.1×10^{-6}
I 130	2.0×10^{-4}	U 238	1.1×10^{-6}
I 131	1.1×10^{-2}	U 239	6.1×10^{-1}
I 132	3.2×10^{-1}	Np 237	4.4×10^{-16}
I 133	3.0×10^{-1}	Np 239	1.5×10^{-3}
I 134	1.0×10^{1}	Np 240	3.5×10^{-11}
I 135	4.6	Pu 236	2.4
		Pu 238	21
		Pu 239	2.1
		Pu 240	3.6
		Pu 241	870
		Pu 242	1.9×10^{-2}

5. RISK ASSESSMENT

For each component of the fuel cycle, and for the source terms associated with the accidents discussed in Section 4, the population dose commitment has been evaluated using the methodology discussed in Section 3.2. For each accident, the critical organ (organ receiving maximum dose) population dose is given together with the population dose to the total body (T.B.). Combining these results with the accident likelihoods also given in Section 4, the expectation value of the population dose commitment is derived and normalized to the annual operation of the generic 1000 MWe LWR using the mass flow factors given in Section 2.2. The normalized population dose commitments in man-rem are then converted to normalized health risks (somatic effects) using the methodology discussed in Section 3.3. All of these results are presented in Tables 5-1 through 5-8 for each component of the supporting LWR fuel cycle.

5.1 Milling

The results for the accidents considered in milling are given in Table 5-1. The consequences and risks associated with the tailings slurry release dominate the accidents from this component of the fuel cycle. As pointed out in Section 4.2, however, since the likelihood of mill tailings dike failure was obtained from historical data, and since the construction techniques for these dikes have been improved, these results probably provide an overestimate of the current risk.

TABLE 5-1

ENVIRONMENTAL RISKS FROM ACCIDENTS IN URANIUM MILLING

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-year)$^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-year (man-rem)	Health Risk per 1000 MWe-year (# of excess cancers)
B.1 Fire in Solvent Extraction Circuit	1.6 (lung) 1.0×10^{-1} (T.B.)	3×10^{-3} to 4×10^{-4}	4.8×10^{-3} to 6.4×10^{-4} 3.0×10^{-4} to 4.0×10^{-5}	6.0×10^{-4} to 8.0×10^{-5} 3.8×10^{-5} to 5.0×10^{-6}	3.8×10^{-8} to 5.0×10^{-9}
E.1 Release of Tailings Slurry from Tailings Pond	2.9 (bone) 1.9×10^{-1} (T.B.)	$\sim 4 \times 10^{-2}$	1.2×10^{-1} 7.6×10^{-3}	1.5×10^{-2} 9.5×10^{-4}	5.5×10^{-7}
E.2 Release of Tailings Slurry from Tailings Distribution Pipeline	1.3×10^{-1} (bone) 8.3×10^{-3} (T.B.)	$\sim 1 \times 10^{-2}$	1.3×10^{-3} 8.3×10^{-5}	1.6×10^{-4} 1.0×10^{-5}	5.8×10^{-9}
Totals (bone) (T.B.)			1.2×10^{-1} 8.0×10^{-3} to 7.7×10^{-3}	1.5×10^{-2} 1.0×10^{-3} to 9.7×10^{-4}	5.9×10^{-7} to 5.6×10^{-7}

5.2 UF_6 Conversion

The results for accidents considered in uranium hexafluoride conversion are given in Table 5-2. The highest consequence accidents are explosions in the uranyl nitrate evaporator or in the hydrogen reduction step of the operation. The risk from the latter accident predominates, because of the higher estimated probability range. If the high end of the estimate is appropriate, the risk from a hydrogen explosion in the reduction step dominates the risk from accidents in uranium hexafluoride conversion. If the low end of the probability range is more correct, the risk from accidents in uranium hexafluoride conversion is dominated by accidents involving the release of UF_6 from cylinders or piping/valve failures in distillation. The probabilities of these events are based upon historical data, and considerable attention has been given in recent years to reducing such releases to a practical minimum.

TABLE 5-2

ENVIRONMENTAL RISKS FROM ACCIDENTS IN URANIUM HEXAFLOURIDE CONVERSION

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood $(\text{plant-yr.})^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
A.1 Uranyl Nitrate Evaporator Explosion	720 (lung) 4.0 (T.B.)	10^{-3} to 10^{-4}	7.2×10^{-1} to 7.2×10^{-2} 4.0×10^{-3} to 4.0×10^{-4}	1.7×10^{-2} to 1.7×10^{-3} 9.5×10^{-5} to 9.5×10^{-6}	7.1×10^{-7} to 7.1×10^{-8}
A.2 Hydrogen Explosion in Reduction	720 (lung) 4.0 (T.B.)	5×10^{-2} to 10^{-3}	36 to 7.2×10^{-1} 2.0×10^{-1} to 4.0×10^{-3}	8.6×10^{-1} to 1.7×10^{-2} 4.8×10^{-3} to 9.5×10^{-5}	3.6×10^{-5} to 7.1×10^{-7}
B.1 Fire in Solvent Extraction Operation	6.2 (lung) 3.9×10^{-1} (T.B.)	$\sim 4 \times 10^{-4}$	2.5×10^{-3} 1.6×10^{-4}	6.0×10^{-5} 3.8×10^{-6}	3.8×10^{-9}
C.1 Release from a Hot UF_6 Cylinder	79 (lung) 4.3×10^{-1} (T.B.)	$\sim 3 \times 10^{-2}$	2.4 1.3×10^{-2}	5.7×10^{-2} 3.1×10^{-4}	2.4×10^{-6}
C.2 Valve Rupture in Distillation Step	29 (lung) 1.6×10^{-1} (T.B.)	$\sim 5 \times 10^{-2}$	1.5 8.0×10^{-3}	3.6×10^{-2} 1.9×10^{-4}	1.5×10^{-6}
E.1 Release of Raffinate from Waste Retention Pond	3.7 (bone) 3.1×10^{-1} (T.B.)	$\sim 2 \times 10^{-2}$	7.4×10^{-2} 6.2×10^{-3}	1.8×10^{-3} 1.5×10^{-4}	8.0×10^{-8}
Totals (lung) (T.B.)			41 to 4.7 2.3 to 10^{-1} to 3.2×10^{-2}	9.7×10^{-1} to 1.1×10^{-1} 5.6×10^{-3} to 7.6×10^{-4}	4.1×10^{-5} to 4.8×10^{-6}

5.3 Enrichment

The results for accidents considered in enrichment are given in Table 5-3. The highest consequence accident postulated in enrichment is the catastrophic fire, for which the upper range of the estimated likelihood is based upon incidents on record (the source term associated with this accident, however, is a rough estimate). However, should the lower range, based upon general chemical industry data, be a more appropriate estimate for the future, the risk associated with the catastrophic fire is relatively insignificant. In this case the release from a hot UF_6 cylinder dominates the risk, and the data used in the assessment of this accident is based upon incidents on record. However, an examination of the historical data reveals that the magnitude of the release associated with this category of accidents has been decreasing over the years.

TABLE 5-3

ENVIRONMENTAL RISKS FROM ACCIDENTS IN ENRICHMENT

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood $(\text{plant-yr.})^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
B.1 Catastrophic Fire	930 (lung) 4.9 (T.B.)	3×10^{-2} to 4×10^{-4}	28 to 3.7×10^{-1} 1.5×10^{-1} to 2.0×10^{-3}	2.2×10^{-1} to 2.9×10^{-3} 1.2×10^{-3} to 1.6×10^{-5}	9.2×10^{-6} to 1.2×10^{-7}
C.1 Release from a Hot UF_6 Cylinder	150 (lung) 7.5×10^{-1}	$\sim 4 \times 10^{-1}$	64 3.0×10^{-1}	5.1×10^{-1} 2.4×10^{-3}	2.1×10^{-5}
C.2 Leaks or Failure of Valves or Piping	1.4 (lung) 7.7×10^{-3} (T.B.)	~ 1.8	2.5 1.4×10^{-2}	2.0×10^{-2} 1.1×10^{-4}	8.4×10^{-7}
D.1 Criticality	4.6×10^{-1} (thyroid) 1.2×10^{-2} (T.B.)	$\sim 8 \times 10^{-5}$	3.7×10^{-5} 9.6×10^{-7}	2.9×10^{-7} 7.6×10^{-9}	2.1×10^{-11}
Total (lung) (T.B.)			95 to 67 4.6×10^{-1} to 3.2×10^{-1}	7.5×10^{-1} to 5.3×10^{-1} 3.7×10^{-3} to 2.5×10^{-3}	3.1×10^{-5} to 2.2×10^{-5}

5.4 Uranium Fuel Fabrication

The results for accidents considered in uranium fuel fabrication are given in Table 5-4. The range of a factor of 10^{-3} in most of the source terms reflects the variability in the design of building ventilation systems in uranium fuel fabrication plants. The highest consequence accident in plants with no building exhaust HEPA filter is the postulated major facility fire, whereas criticality is the highest consequence accident in plants equipped with a building exhaust HEPA filter. Because of the considerably higher probability associated with the release from a hot UF_6 cylinder, the risk associated with this accident dominates the total risk from uranium fuel fabrication. It should be pointed out that the consequences of this particular accident in uranium fuel fabrication (in the absence of a building exhaust HEPA filter) is significantly higher than in other components of the fuel cycle, largely because of the higher population density in the vicinity of the generic uranium fuel fabrication plant.

TABLE 5-4

ENVIRONMENTAL RISKS FROM ACCIDENTS IN URANIUM FABRICATION

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood $(\text{plant-yr.})^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 Mwe-yr. (# of excess cancers)
A.1 Hydrogen Explosion in Reduction Furnace	16 to 1.6×10^{-2} (lung) 7.4×10^{-2} to 7.4×10^{-5} (T.B.)	5×10^{-2} to 2×10^{-3}	8.0×10^{-1} to 3.2×10^{-5} 3.7×10^{-3} to 1.5×10^{-7}	3.2×10^{-2} to 1.3×10^{-6} 1.5×10^{-4} to 6.0×10^{-9}	1.3×10^{-6} to 5.4×10^{-11}
B.1 Major Facility Fire	1.6×10^{4} to 16 (lung) 74 to 7.4×10^{-2}	$\sim 2 \times 10^{-4}$	3.2 to 3.2×10^{-3} 1.5×10^{-2} to 1.5×10^{-5}	1.3×10^{-1} to 1.3×10^{-4} 6.0×10^{-4} to 6.0×10^{-7}	5.4×10^{-6} to 5.4×10^{-9}
B.2 Fire in a Roughing Filter	3.8 to 3.8×10^{-3} (lung) 1.8×10^{-2} to 1.8×10^{-5} (T.B.)	$\sim 10^{-2}$	3.8×10^{-2} to 3.8×10^{-5} 1.8×10^{-4} to 1.8×10^{-7}	1.5×10^{-3} to 1.5×10^{-6} 7.2×10^{-6} to 7.2×10^{-9}	6.3×10^{-8} to 6.3×10^{-11}
C.1 Release from a Hot UF_6 Cylinder	1600 to 1.6 (lung) 7.8 to 7.8×10^{-3} (T.B.)	$\sim 3 \times 10^{-2}$	48 to 4.8×10^{-2} 2.3×10^{-1} to 2.3×10^{-4}	1.9 to 1.9×10^{-3} 9.2×10^{-3} to 9.2×10^{-6}	7.9×10^{-5} to 7.9×10^{-8}
C.2 Failure of Valves or Piping	460 to 4.6×10^{-1} (lung) 2.2 to 2.2×10^{-3} (T.B.)	$\sim 4 \times 10^{-3}$	1.8 to 1.8×10^{-3} 8.8×10^{-3} to 8.8×10^{-6}	7.2×10^{-2} to 7.2×10^{-5} 3.5×10^{-4} to 3.5×10^{-7}	3.0×10^{-6} to 3.0×10^{-9}
D.1 Criticality	32 (thyroid) 1.1 (T.B.)	$\sim 8 \times 10^{-4}$	2.6×10^{-2} 8.8×10^{-4}	1.0×10^{-3} 3.5×10^{-5}	7.4×10^{-8}
E.1 Waste Retention Pond Failure	5.7×10^{-1} (bone) 3.5×10^{-2} (T.B.)	2×10^{-2} to 2×10^{-3}	1.1×10^{-2} to 1.1×10^{-3} 7.0×10^{-4} to 7.0×10^{-5}	4.4×10^{-4} to 4.4×10^{-5} 2.8×10^{-5} to 2.8×10^{-6}	1.6×10^{-8} to 1.6×10^{-9}
Totals (lung) (thyroid) (T.B.)			54 to 5.3×10^{-2} 2.6×10^{-2} 2.6×10^{-1} to 1.2×10^{-3}	2.1 to 2.1×10^{-3} 1.0×10^{-3} 1.0×10^{-2} to 4.8×10^{-5}	8.9×10^{-5} to 1.6×10^{-7}

5.5 Reprocessing

The results for accidents considered in reprocessing facilities are given in Table 5-5. The accident of highest consequence is the postulated explosion in the waste calciner, for which a volatile fraction of ruthenium of 10^{-2} has been postulated because of the high operating temperatures. The consequences of several of the accidents compounded by postulated failures of the exhaust HEPA filters are nearly of the same order of magnitude, however the corresponding likelihoods are exceedingly low. It is of interest to note, in fact, that with the exception of three accidents, the estimated risk associated with multiple HEPA filter failure is lower than that associated with normal filter operation. This result derives from the fact that, in most cases, the volatile fraction of ruthenium dominates the resulting population dose commitment.

One of the three accidents for which the postulated multiple filter failure contributes significantly to the dose is the release in the fuel receiving and storage area, for which the vapors released to the cask were assumed to be converted to the particulate form during passage through the pool. In fact, this accident dominates the risk at the high end of the estimate of likelihood and constitutes about 25% of the total risk from reprocessing at the low end of the range of likelihoods. The ion exchange resin fire contributes substantially to the total risk at the high end of the range of likelihoods, and is insignificant at the low end. Because of the potential importance of both of these accidents, the probabilities of their occurrence should be placed on a firmer foundation. The risk from

TABLE 5-5

ENVIRONMENTAL RISKS FROM ACCIDENTS IN FUEL REPROCESSING

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-yr.) -1	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
A.1 Explosion in High Aqueous Waste Concentration					
a. Normal HEPA Filtration	230,000 (G.I.) 430 (T.B.)	~ 10^{-5}	2.3 4.3×10^{-3}	5.3×10^{-2} 1.0×10^{-4}	3.3×10^{-6}
b. HEPA Filter Failure	3.3×10^5 (G.I.) 9.5×10^3 (T.B.)	~ 10^{-8}	3.3×10^{-3} 9.5×10^{-5}	7.7×10^{-5} 2.2 z 10^{-6}	5.5×10^{-9}
A.2 Explosion in Low Aqueous Waste Concentrator					
a. Normal HEPA Filtration	15,000 (G.I.) 28 (T.B.)	~ 10^{-4}	1.5 2.8×10^{-3}	3.5×10^{-2} 6.5×10^{-5}	2.2×10^{-6}
b. HEPA Filter Failure	1.6×10^4 (G.I.) 4.8×10^1 (T.B.)	~ 10^{-7}	1.6×10^{-3} 4.8×10^{-6}	3.7×10^{-5} 1.1×10^{-7}	2.3×10^{-9}
A.3 Explosion in High Aqueous Feed Tank					
a. Normal HEPA Filtration	840,000 (G.I.) 1,600 (T.B.)	~ 10^{-5}	8.4 1.6×10^{-2}	2.0×10^{-1} 3.7×10^{-4}	1.2×10^{-5}
b. HEPA Filter Failure	8.4×10^5 (G.I.) 1.7×10^3 (T.B.)	~ 10^{-7}	8.4×10^{-2} 1.7×10^{-4}	2.0×10^{-3} 3.8×10^{-6}	1.2×10^{-7}
A 4 Explosion in Waste Calciner					
a. Normal HEPA Filtration	2.3×10^6 (G.I.) 4,300 (T.B.)	~ 10^{-6}	2.3 4.3×10^{-3}	5.3×10^{-2} 1.0×10^{-4}	3.3×10^{-6}
b. HEFA Filter Failure	2.4×10^6 (G.I.) 1.3×10^4 (T.B.)	~ 10^{-9}	2.4×10^{-3} 1.3×10^{-5}	5.6×10^{-5} 3.0×10^{-7}	3.6×10^{-9}

TABLE 5-5

(continued)

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-yr.)$^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
A.5 Explosion in Iodine Adsorber	1.9×10^3 (thyroid) 4.8 (T.B.)	$\sim 2 \times 10^{-4}$	3.8×10^{-1} 9.6×10^{-4}	8.8×10^{-3} 2.2×10^{-5}	5.6×10^{-7}
B.1 Solvent Fire in Codecontamination Cycle					
a. Normal HEPA Filtration	14,000 (G.I.) 23 (T.B.)	10^{-4} to 10^{-6}	1.4 to 1.4×10^{-2} 2.3×10^{-3} to 2.3×10^{-5}	3.3×10^{-2} to 3.3×10^{-4} 5.3×10^{-5} to 5.3×10^{-7}	2.0×10^{-6} to 2.0×10^{-8}
b. HEPA Filter Failure	1.5×10^4 (G.I.) 5.6×10^1 (T.B.)	10^{-7} to 10^{-9}	1.5×10^{-3} to 1.5×10^{-5} 5.6×10^{-6} to 5.6×10^{-8}	3.5×10^{-5} to 3.5×10^{-7} 1.3×10^{-7} to 1.3×10^{-9}	2.2×10^{-9} to 2.2×10^{-11}
B.2 Solvent Fire in Plutonium Extraction Cycle					
a. Normal HEPA Filtration	1.5×10^{-2} (bone) 3.1×10^{-4} (T.B.)	10^{-4} to 10^{-6}	1.5×10^{-6} to 1.5×10^{-8} 3.1×10^{-8} to 3.1×10^{-10}	3.5×10^{-8} to 3.5×10^{-10} 7.2×10^{-10} to 7.2×10^{-12}	6.6×10^{-13} to 6.6×10^{-15}
b. HEPA Filter Failure	2.6×10^4 (bone) 5.2×10^2 (T.B.)	10^{-9} to 10^{-11}	2.6×10^{-5} to 2.6×10^{-7} 5.2×10^{-7} to 5.2×10^{-9}	6.0×10^{-7} to 6.0×10^{-9} 1.2×10^{-8} to 1.2×10^{-10}	4.1×10^{-11} to 4.1×10^{-13}
B.3 Ion-Exchange Resin Fire					
a. Normal HEPA Filtration	190 (G.I.) 3.6×10^{-1} (T.B.)	10^{-1} to 10^{-4}	19 to 1.9×10^{-2} 3.6×10^{-2} to 3.6×10^{-5}	4.4×10^{-1} to 4.4×10^{-4} 8.4×10^{-4} to 8.4×10^{-7}	2.7×10^{-5} to 2.7×10^{-8}
b. HEPA Filter Failure	8.3×10^4 (bone) 1.8×10^3 (T.B.)	10^{-6} to 10^{-9}	8.3×10^{-2} to 8.3×10^{-5} 1.8×10^{-3} to 1.8×10^{-6}	1.9×10^{-3} to 1.9×10^{-6} 4.2×10^{-5} to 4.2×10^{-8}	1.3×10^{-7} to 1.3×10^{-10}

TABLE 5-5

(continued)

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-yr.) -1	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
C.1 Fuel Assembly Rupture and Release in Fuel Receiving and Storage					
a. Normal HEPA Filtration	6.8 (G.I.) 1.3×10^{-2} (T.B.)	10^{-1} to 10^{-2}	6.8×10^{-1} to 6.8×10^{-2} 1.3×10^{-3} to 1.3×10^{-4}	1.6×10^{-2} to 1.6×10^{-3} 3.0×10^{-5} to 3.0×10^{-6}	1.0×10^{-6} to 1.0×10^{-7}
b. HEPA Filter Failure	6.8×10^{5} (G.I.) 1.3×10^{3} (T.B.)	10^{-4} to 10^{-5}	6.8×10^{1} to 6.8 1.3×10^{-1} to 1.3×10^{-2}	1.6 to 1.6×10^{-1} 3.0×10^{-3} to 3.0×10^{-4}	1.0×10^{-4} to 1.0×10^{-5}
C.2 Dissolver Seal Failure					
a. Normal HEPA Filtration	1.6×10^{-1} (lung) 2.3×10^{-2} (T.B.)	$\sim 10^{-5}$	1.6×10^{-6} 2.3×10^{-7}	3.7×10^{-8} 5.3×10^{-9}	3.4×10^{-12}
b. HEPA Filter Failure	1.6×10^{4} (lung) 2.3×10^{3} (T.B.)	$\sim 10^{-8}$	1.6×10^{-4} 2.3×10^{-5}	3.7×10^{-6} 5.3×10^{-7}	3.4×10^{-10}
C.3 Release from a Hot UF_6 Cylinder	3.2×10^{2} (lung) 1.5 (T.B.)	$\sim 5 \times 10^{-2}$	1.6×10^{1} 7.5×10^{-2}	3.7×10^{-1} 1.7×10^{-3}	1.5×10^{-5}
D.1 Criticality					
a. Normal HEPA Filtration	3.0×10^{-2} (T.B.)	8×10^{-3} to 3×10^{-5}	2.4×10^{-4} to 9.0×10^{-7}	5.6×10^{-6} to 2.1×10^{-8}	2.2×10^{-9} to 8.4×10^{-12}
b. HEPA Filter Failure	2.5×10^{-1} (lung) 3.5×10^{-2} (total body)	8×10^{-6} to 3×10^{-8}	2.0×10^{-6} to 7.5×10^{-9} 2.8×10^{-7} to 1.1×10^{-9}	4.7×10^{-8} to 1.7×10^{-10} 6.5×10^{-9} to 2.6×10^{-11}	4.2×10^{-12} to 1.6×10^{-14}
Totals (G.I.) (lung) (T.B.)			1.0×10^{2} to 2.1×10^{1} 16 2.7×10^{-1} to 1.2×10^{-1}	2.4 to 5.0×10^{-1} 3.7×10^{1} 6.3×10^{-3} to 2.8×10^{-3}	1.7×10^{-4} to 4.7×10^{-5}

all postulated explosions in reprocessing constitute roughly 1/8 of the total risk at the high end of the range of likelihoods and as much as 1/2 at the low end. However, the likelihoods estimated for explosions in reprocessing are at least one order of magnitude, and, possibly as high as two orders of magnitude smaller than corresponding estimates for explosion likelihoods in other industries. Further work in this area would contribute more confidence to these results.

5.6 Mixed Oxide Fuel Fabrication

The results for accidents considered in mixed oxide fuel fabrication are given in Table 5-6. The consequences and risks associated with postulated multiple failures of the exhaust HEPA filters dominate the total risk from this component of the fuel cycle. This is because, with the exception of criticality, the sources are particulate and the failure of the two building exhaust HEPA filters is estimated to increase the release to the environment by a factor of 10^5 (whereas the probability of simultaneous failure of the dual exhaust HEPA filters is estimated to be 10^{-3}/demand). Although the risks associated with a few of the accidents are relatively insignificant (glove failure and criticality), several of the remaining accidents contribute in roughly equal amounts to the total risk from accidents in mixed oxide fuel fabrication.

TABLE 5-6

ENVIRONMENTAL RISKS FROM ACCIDENTS IN MIXED OXIDE FUEL FABRICATION

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood $(\text{plant-yr.})^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
A.1 Explosion in Oxidation-Reduction Scrap Furnace					
a. Normal HEPA Filtration	1.5 (bone) 3.1×10^{-2} (T.B.)	5×10^{-2} to 2×10^{-3}	7.5×10^{-2} to 3.0×10^{-3} 1.6×10^{-3} to 6.2×10^{-5}	3.3×10^{-3} to 1.3×10^{-4} 7.0×10^{-5} to 2.7×10^{-6}	6.7×10^{-8} to 2.6×10^{-9}
b. HEPA Filter Failure	1.5×10^{5} (bone) 3.1×10^{3} (T.B.)	5×10^{-5} to 2×10^{-6}	7.5 to 3.0×10^{-1} 1.6×10^{-1} to 6.2×10^{-3}	3.3×10^{-1} to 1.3×10^{-2} 7.0×10^{-3} to 2.7×10^{-4}	6.7×10^{-6} to 2.6×10^{-7}
B.1 Major Facility Fire					
a. Normal HEPA Filtration	7.6×10^{1} (bone) 1.6 (T.B.)	$\sim 2 \times 10^{-4}$	1.5×10^{-2} 3.2×10^{-4}	6.5×10^{-4} 1.4×10^{-5}	1.3×10^{-8}
b. HEPA Filter Failure	7.6×10^{6} (bone) 1.4×10^{5} (T.B.) (plus 25 short-term deaths)	$\sim 2 \times 10^{-7}$	1.5 2.8×10^{-2}	6.5×10^{-2} 1.2×10^{-3}	1.2×10^{-6} (plus 2.2×10^{-7} short-term deaths)
B.2 Fire in Waste Compaction Glove Box					
a. Normal HEPA Filtration	1.5×10^{-1} (bone) 3.1×10^{-3} (T.B.)	$\sim 10^{-2}$	1.5×10^{-3} 3.1×10^{-5}	6.5×10^{-5} 1.3×10^{-6}	1.3×10^{-9}
b. HEPA Filter Failure	1.5×10^{4} (bone) 3.1×10^{2} (T.B.)	$\sim 10^{-5}$	1.5×10^{-1} 3.1×10^{-3}	6.5×10^{-3} 1.3×10^{-4}	1.3×10^{-7}
B.3 Ion-Exchange Resin Fire					
a. Normal HEPA Filtration	4.5×10^{-1} (bone) 9.2×10^{-3} (T.B.)	10^{-1} to 10^{-4}	4.5×10^{-2} to 4.5×10^{-5} 9.2×10^{-4} to 9.2×10^{-7}	2.0×10^{-3} to 2.0×10^{-6} 4.0×10^{-5} to 4.0×10^{-8}	4.0×10^{-8} to 4.0×10^{-11}
b. HEPA Filter Failure	4.5×10^{4} (bone) 9.2×10^{2} (T.B.)	10^{-4} to 10^{-7}	4.5 to 4.5×10^{-3} 9.2×10^{-2} to 9.2×10^{-5}	2.0×10^{-1} to 2.0×10^{-4} 4.0×10^{-3} to 4.0×10^{-6}	4.0×10^{-6} to 4.0×10^{-9}

TABLE 5-6

(continued)

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-yr.) $^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
B.4 Dissolver Fire in Scrap Recovery					
a. Normal HEPA Filtration	7.6 (bone) 1.6×10^{-1} (T.B.)	$\sim 10^{-2}$	7.6×10^{-2} 1.6×10^{-3}	3.3×10^{-3} 7.0×10^{-5}	6.7×10^{-8}
b. HEPA Filter Failure	7.6×10^{5} (bone) 1.6×10^{4} (T.B.)	$\sim 10^{-5}$	7.6 1.6×10^{-1}	3.3×10^{-1} 7.0×10^{-3}	6.7×10^{-6}
C.1 Glove Failure					
a. Normal HEPA Filtration	6.1×10^{-4} (bone) 1.3×10^{-5} (T.B.)	~ 1	6.1×10^{-4} 1.3×10^{-5}	2.7×10^{-5} 5.7×10^{-7}	5.4×10^{-10}
b. HEPA Filter Failure	6.1×10^{1} (bone) 1.3 (T.B.)	$\sim 10^{-3}$	6.1×10^{-2} 1.3×10^{-3}	2.7×10^{-3} 5.7×10^{-5}	5.4×10^{-8}
C.2 Severe Glove Box Damage					
a. Normal HEPA Filtration	3.0 (bone) 6.1×10^{-2} (T.B.)	$\sim 10^{-2}$	3.0×10^{-2} 6.1×10^{-4}	1.3×10^{-3} 2.7×10^{-5}	2.6×10^{-8}
b. HEPA Filter Failure	3.0×10^{5} (bone) 6.1×10^{3} (T.B.)	$\sim 10^{-5}$	3.0 6.1×10^{-2}	1.3×10^{-1} 2.7×10^{-3}	2.6×10^{-6}
D.1 Criticality					
a. Normal HEPA Filtration	13 (thyroid) 3.8×10^{-1} (T.B.)	8.0×10^{-3} to 3.0×10^{-5}	1.0×10^{-1} to 3.9×10^{-4} 3.0×10^{-3} to 1.1×10^{-5}	4.3×10^{-3} to 1.7×10^{-5} 1.3×10^{-4} to 4.8×10^{-7}	3.1×10^{-7} to 1.2×10^{-9}
b. HEPA Filter Failure	2.0×10^{4} (bone) 4.2×10^{2} (T.B.)	8.0×10^{-6} to 3.0×10^{-8}	1.6×10^{-1} to 6.0×10^{-4} 3.4×10^{-3} to 1.3×10^{-5}	7.0×10^{-3} to 2.6×10^{-5} 1.5×10^{-4} to 5.7×10^{-7}	1.4×10^{-7} to 5.3×10^{-10}
Totals (bone) (thyroid) (T.B.)			2.5×10^{1} to 1.3×10^{1} 1.0×10^{-1} to 3.9×10^{-4} 5.2×10^{-1} to 3.2×10^{-1}	1.1 to 5.5×10^{-1} 4.3×10^{-3} to 1.7×10^{-5} 1.9×10^{-2} to 1.2×10^{-2}	2.2×10^{-5} to 1.1×10^{-5} (plus 2.2×10^{-7} short-term deaths)

5.7 Plutonium Storage

The results for the criticality accident considered in plutonium storage are given in Table 5-7. Bone is the critical organ for this release, because a relatively small number of fissions is postulated (10^{17}) and a small fraction of the plutonium in a container is assumed to be released by the energy evolved in the excursion. Accordingly, the risk associated with the postulated multiple failure in the exhaust HEPA filters dominates the normal HEPA operation case. The risk associated with this accident is in the range of the other postulated criticalities in the fuel cycle, but is, nonetheless, relatively insignificant.

TABLE 5-7

ENVIRONMENTAL RISKS FROM ACCIDENTS IN PLUTONIUM STORAGE

Accident	Population Dose for Generic Plant (man-rem)	Accident Likelihood (plant-yr.) -1	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
D.1 Criticality					
a. Normal HEPA Filtration	2.8 (bone) 8.8×10^{-1} (T.B.)	~ 8×10^{-5}	2.2×10^{-4} 7.0×10^{-5}	3.9×10^{-7} 1.2×10^{-7}	5.2×10^{-11}
b. HEPA Filter Failure	2.8×10^{5} (bone) 5.7×10^{3} (T.B.)	~ 8×10^{-8}	2.2×10^{-2} 4.6×10^{-4}	3.9×10^{-5} 8.1×10^{-7}	7.8×10^{-10}
Totals (bone) (T.B.)			2.2×10^{-2} 5.3×10^{-4}	3.9×10^{-5} 9.3×10^{-7}	8.3×10^{-10}

5.8 Transportation

The results for the accidents considered in transportation are given in Table 5-8. The highest consequence accident is the postulated criticality in PuO_2 transport, for which 1% of the plutonium or approximately 60 grams, is assumed to be expelled from one of the containers from the energy released in the excursion. This estimate is based upon little corroborative data; however, the risk associated with this accident is relatively insignificant. The predicted highest risk accident is the collision involving irradiated fuel. However, the uncertainty in this estimate is large, as seen by the range in the predicted health risk. The predicted risk associated with shipments of UF_6 cylinders is also relatively significant. It should be noted, however, that as there is essentially no data base from which to evaluate potential releases from severe transportation accidents involving radioactive materials, the estimates contained here are highly uncertain.

TABLE 5-8

ENVIRONMENTAL RISKS FROM ACCIDENTS IN TRANSPORTATION

Accident	Population Dose for Generic Shipment (man-rem)	Accident Likelihood (shipment)$^{-1}$	Population Dose Expectation Value (man-rem)	Population Dose per 1000 MWe-yr. (man-rem)	Health Risk per 1000 MWe-yr. (# of excess cancers)
1. Leakage of coolant from irradiated fuel cask	7.2×10^{-4} (bone) 5.8×10^{-4} (T.B.)	$\sim 3 \times 10^{-4}$	2.2×10^{-7} 1.8×10^{-7}	2.0×10^{-6} 1.7×10^{-6}	6.8×10^{-10}
2. Improperly closed plutonium oxide container	56 (bone) 1.1 (T.B.)	1×10^{-3} to 4×10^{-4}	5.6×10^{-2} to 2.2×10^{-2} 1.1×10^{-3} to 4.4×10^{-4}	1.6×10^{-1} to 6.2×10^{-2} 3.1×10^{-3} to 1.2×10^{-3}	3.1×10^{-6} to 1.2×10^{-6}
3. Release from a collision involving natural UF_6	38,000 (lung) 200 (T.B.)	$\sim 4 \times 10^{-6}$	1.5×10^{-1} 8.0×10^{-4}	1.6 8.4×10^{-3}	6.7×10^{-5}
4. Release from a collision involving enriched UF_6	1.4×10^{5} (lung) 660 (T.B.)	$\sim 6 \times 10^{-6}$	8.4×10^{-1} 4.0×10^{-3}	2.6 1.2×10^{-2}	1.1×10^{-4}
5. Release from a collision involving irradiated fuel	190,000 (G.I.) 19,000 (T.B.)	9×10^{-6} to 2×10^{-8}	1.7 to 3.8×10^{-3} 1.7×10^{-1} to 3.8×10^{-4}	16 to 3.5×10^{-2} 1.6 to 3.5×10^{-3}	1.5×10^{-3} to 3.4×10^{-6}
6. Release from a collision involving irradiated fuel followed by release of fuel from the cask	200,000 (G.I.) 27,000 (T.B.) (plus 14 short-term deaths)	9×10^{-8} to 2×10^{-10}	1.8×10^{-2} to 4.0×10^{-5} 2.4×10^{-3} to 5.4×10^{-6}	1.7×10^{-1} to 3.7×10^{-4} 2.2×10^{-2} to 5.0×10^{-5}	1.8×10^{-5} to 4.0×10^{-8} (plus 1.2×10^{-5} to 2.6×10^{-8} short-term deaths)
7. Release from a collision involving plutonium oxide	7.0×10^{4} (bone) 1.4×10^{3} (T.B.)	3×10^{-6} to 2×10^{-9}	2.1×10^{-1} to 1.4×10^{-4} 4.2×10^{-3} to 2.8×10^{-6}	5.8×10^{-1} to 3.9×10^{-4} 1.2×10^{-2} to 7.8×10^{-6}	1.2×10^{-5} to 7.7×10^{-9}
8. Criticality of unirradiated fuel	3.8 (T.B.)	5×10^{-8} to 2×10^{-10}	1.9×10^{-7} to 7.6×10^{-10}	1.4×10^{-6} to 5.7×10^{-9}	5.6×10^{-10} to 2.3×10^{-12}
9. Criticality of enriched UO_2	11.6 (thyroid) 4.0 (T.B.)	8×10^{-8} to 5×10^{-11}	9.3×10^{-7} to 5.8×10^{-10} 3.2×10^{-7} to 2.0×10^{-10}	5.3×10^{-6} to 3.3×10^{-9} 1.8×10^{-6} to 1.1×10^{-9}	9.4×10^{-10} to 5.8×10^{-13}
10. Criticality of PuO_2	1.2×10^{6} (bone) 25,000 (T.B.)	3×10^{-8} to 2×10^{-11}	3.6×10^{-2} to 2.4×10^{-5} 7.5×10^{-4} to 5.0×10^{-7}	1.0×10^{-1} to 6.6×10^{-5} 2.1×10^{-3} to 1.4×10^{-6}	2.0×10^{-6} to 1.3×10^{-9}
Total				4.2 (lung) 16 to .035 (G.I.) 1.7 to 2.5×10^{-2} (T.B.)	1.7×10^{-3} to 1.8×10^{-4} (plus 1.2×10^{-5} to 2.6×10^{-8} short-term deaths)

5.9 Overall Fuel Cycle Risks and Comparisons

The total risks from the accidents considered in each component of the supporting LWR fuel cycle are summarized and aggregated in Table 5-9. Because of the uncertainties associated with the linear, non-threshold dose-response relationship, population dose is given in addition to somatic health effects. Genetic effects are even more uncertain than somatic effects; however, based upon the correlation given in Section 3.3 from Reference 12, the number of predicted congenital defects resulting from accidents in the LWR supporting fuel cycle is in the range of 5.4×10^{-4} to 1.3×10^{-5} per 1000 MWe-year, nearly an order of magnitude lower than the somatic risk.

Transportation accidents dominate the total accident risk, whereas the risk from accidents in mining, milling, and plutonium storage are relatively insignificant. The risks from the remaining components of the fuel cycle, with the exception of spent fuel reprocessing, are roughly of equal orders of magnitude, although the risk from uranium fuel fabrication is negligible at the low end of the predicted range, characterized by the existence of a HEPA filter in the building exhaust system.

It is of interest to note that despite the relatively large range of values given for a number of individual fuel cycle components, the ranges in the totals, with the exception of uranium fuel fabrication, are relatively small.

TABLE 5-9

TOTAL ENVIRONMENTAL HEALTH RISKS FROM ACCIDENTS IN THE LWR SUPPORTING FUEL CYCLE

Fuel Cycle Component	Population Dose per 1000 MWe-yr. (man-rem)	Somatic Health Risk per 1000 MWe-yr. (# of excess cancers)
Uranium Mining	0	0
Uranium Milling	.015 (bone) .001 (T.B.)	5.9×10^{-7} to 5.6×10^{-7}
UF_6 Conversion	.97 to .11 (lung) .0056 to .00076 (T.B.)	4.1×10^{-5} to 4.8×10^{-6}
Enrichment	.75 to .53 (lung) .0037 to .0025 (T.B.)	3.1×10^{-5} to 2.2×10^{-5}
Uranium Fuel Fabrication	2.1 to .0021 (lung) .010 to 4.8×10^{-5} (T.B.)	8.9×10^{-5} to 1.6×10^{-7}
Reprocessing	.37 (lung) 2.4 to .50 (G.I.) .0063 to .0028 (T.B.)	1.7×10^{-4} to 4.7×10^{-5}
Mixed Oxide Fabrication	1.1 to .55 (bone) .019 to .012 (T.B.)	2.2×10^{-5} to 1.1×10^{-5}
Plutonium Storage	3.9×10^{-5} (bone) 9.3×10^{-7} (T.B.)	8.3×10^{-10}
Transportation	4.2 (lung) 16 to .035 (G.I.) 1.7 to .025 (T.B.)	1.7×10^{-3} to 1.8×10^{-4}
Totals	8.4 to 5.2 (lung) 18 to .54 (G.I.) 1.1 to .57 (bone) 1.8 to .044 (T.B.)	2.1×10^{-3} to 2.7×10^{-4}

The uncertainty in the results for total estimated health risk is composed of three main components. These are 1) the aggregated uncertainty associated with the estimated source terms and accident likelihoods broken down in Section 4; 2) the uncertainty associated with the dose conversion model discussed in Section 3.2; and 3) the uncertainty associated with the dose-response model discussed in Section 3.3. Although an estimate for a source term and likelihood associated with any particular accident may be in error by several orders of magnitude, it is estimated that, for the class of accidents considered in this study, the range of uncertainty in the aggregated expectation value of consequences in the total supporting fuel cycle is roughly an order of magnitude, with a reasonable degree of confidence.

The range of uncertainty associated with the dose conversion may also be as high as an order of magnitude, but considering the degree of conservatism factored into these models, the results of the dose conversion are most likely biased toward the high side. The combined uncertainty from the source term/likelihood estimates and from dose conversion, may be as high as a factor of 10 to 100, and the combined results are probably also biased toward the high side.

The conversion to health effects using the linear, non-threshold dose-response hypothesis is also highly uncertain. The degree of realism associated with this hypothesis is the subject of a continuing debate within the scientific community. It is generally held that the model is conservative; however, no attempt will be made here to speculate on the extent of uncertainty associated with its application.

Our estimates do not include the risk associated with certain "class 9" accidents, in particular site-related events such as tornados, hurricanes, floods, or missile impacts. The risk associated with these events has been dismissed in comparison with process initiated events in earlier studies.[9] Moreover, this study does not address the risks associated with the management of nuclear wastes, including the storage of high level wastes as liquids or solids at the reprocessing facility.

The results given in this section are normalized to the LWR fuel cycle incorporating the recycle of plutonium. Should plutonium not be recycled, the LWR fuel cycle would be altered as shown in Figure 2-1 and quantified in Tables 2-5 and 2-6 (assuming the maintenance of reprocessing and recycle of recovered uranium). Renormalizing to the non-recycle case, and taking into account the slightly altered source term from reprocessing, the total risk given in Table 5-9 would be relatively unchanged at the high end of the range and increased by roughly 30% at the low end in the absence of plutonium recycle. This result stems from the increase in the required amount of UF_6 conversion, enrichment, and transportation in the absence of plutonium recycle, which compensates for the reduction in risk from mixed oxide fuel fabrication and shipments of spent fuel.

The risks from accidents in the LWR supporting fuel cycle are compared with those associated with normal operations in Table 5-10. The normal operations source terms used in this evaluation are compiled in Appendix B. The source terms were converted to population dose and health risk using the same methodology adopted for accidents, and as discussed in Section 3.

TABLE 5-10

COMPARISON BETWEEN ENVIRONMENTAL HEALTH RISKS FROM ACCIDENTS AND FROM NORMAL OPERATIONS OF THE LWR FUEL CYCLE

	Risks from Normal Operations		Risks from Accidents
Fuel Cycle Component	Population Dose per 1000 MWe-year (man-rem)	Health Risk per 1000 MWe-year (# of excess cancers)	Health Risk per 1000 MWe-year (# of excess cancers)
Uranium Mining	1.4×10^3 (lung) 2.2×10^3 (bone) 7.2×10^2 (T.B.)	3.3×10^{-1}	0
Uranium Milling	4.0×10^3 (lung) 4.4×10^3 (bone) 1.7×10^3 (T.B.)	8.0×10^{-1}*	5.9×10^{-7} to 5.6×10^{-7}
UF_6 Conversion	0.81 (lung) 2.0×10^{-2} (T.B.)	3.9×10^{-5}	4.1×10^{-5} to 4.8×10^{-6}
Enrichment	1.1 (lung) 1.9×10^{-2} (T.B.)	5.1×10^{-5}	3.1×10^{-5} to 2.2×10^{-5}
Uranium Fuel Fabrication	1.3 (lung) 6.2×10^{-3} (T.B.)	5.4×10^{-5}	8.9×10^{-5} to 1.6×10^{-7}
Reprocessing	1500 (thyroid) 790 (T.B.)	3.6×10^{-1}**	1.7×10^{-4} to 4.7×10^{-5}
Mixed Oxide Fabrication	2.7 (bone) .057 (T.B.)	5.5×10^{-5}	2.2×10^{-5} to 1.1×10^{-5}
Plutonium Storage	0	0	8.3×10^{-10}
Transportation	0.35 (T.B.)	1.4×10^{-4}	1.7×10^{-3} to 1.8×10^{-4}
Totals	1500 (thyroid) 5400 (lung) 3200 (T.B.)	1.5	2.1×10^{-3} to 2.7×10^{-4}
Reactor	36 (thyroid) 0.94 (T.B.)	2.5×10^{-3}	

* Control of the tailings pile (covering the pile after the mill has been shut down) would reduce this value to 7.3×10^{-2}

** Control of C-14 emissions to 1% of normal release coupled with proposed EPA radiation protection control on Kr-85, I-129 and plutonium would reduce this value to 1.3×10^{-2}.

It is seen that the total health risk from the accidents considered in this study is orders of magnitude lower than the health risk associated with normal operations of the supporting fuel cycle, and comparable or lower than the health risk associated with normal operation of the reactor. The principal contributors to the risk associated with normal operations of the supporting fuel cycle are the mining, milling, and reprocessing components. Additional controls on the emissions from reprocessing and milling could reduce the risk from normal operations of the supporting fuel cycle by roughly a factor of four. Still the risk from accidents would be negligible in comparison.*

It is, however, interesting to note that, with the exception of mining, milling, reprocessing and transportation, the risk from accidents in each component of the supporting fuel cycle is roughly comparable or possibly larger than from normal operations. For these fuel cycle components, then, the results of this study indicate that accidents have the effect of increasing the risks associated with normal operations by roughly a factor of two.

The estimates of health risks from both normal operations and from accidents compared in Table 5-10 are made on the basis of the same dose conversion models and dose-response relationship. On a relative basis, then, the uncertainty in this comparison derives primarily from uncertainties in the normal operations source terms and the expectation values of the accident consequences. Assuming that the former contributes insignificantly to the overall uncertainty, the range of uncertainty in the comparison is estimated to be roughly an order of magnitude.

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* This conclusion does not take into consideration the theoretical increase in the risk of accidents resulting from potential releases associated with these new control measures.

The risks associated with reactor accidents have been evaluated by the Rasmussen study.[6] The preliminary results of this study indicate that the approximate societal risks associated with the annual operation of a generic 1000 MWe LWR consist of 4×10^{-4} acute fatalities, 8×10^{-4} acute illnesses, and 3×10^{-3} latent cancers.* This places the somatic health risk from reactor accidents slightly in excess of both the risk from normal reactor operation and the risk associated with accidents in the supporting fuel cycle.

It is also of interest to compare these results with estimates of occupational health risks associated with the generation of nuclear electric power. Reference 52 compares occupational health risks across the various fuel cycles. A total occupational health risk of 8×10^{-2} malignancies is attributed to the annual operation of a generic LWR. This is further subdivided into 1×10^{-2} occupational malignancies from uranium mining and 7×10^{-2} malignancies from all other fuel cycle steps.* These risks are also well in excess of the estimated risk to the general population from accidents in the supporting fuel cycle.

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* This latent cancer estimate is based upon a linear dose-response conversion factor of 100 cancers per 10^6 man-rem, lower by at least a factor of four than the conversion factor used in this study.

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* These estimates are based upon linear dose-response conversion factors of 10^{-3} lung cancers per miner - WLM and 200×10^{-6} malignancies per man-rad. This latter number is at least a factor of two lower than the conversion factor used in this study.

6. CONCLUSIONS AND RECOMMENDATIONS FOR FUTURE WORK

The somatic health risk associated with accidents in the fuel cycle supporting the annual operation of a 1000 MWe LWR is estimated to be of the order of roughly 10^{-3} excess cancers. This result is synthesized from nominal radiological source terms and accident likelihood data compiled from a number of diverse sources and subjected to interpretation, renormalization, and revision.

The estimate is subject to uncertainties associated with the dose conversion model and the dose-response relationship, in addition to the vagaries of the consequence expectation value estimate. Nevertheless, assuming the validity of the linear, non-threshold dose-response hypothesis, the uncertainty in the aggregate risk estimate is considered to be roughly one to two orders of magnitude, with a reasonable degree of confidence.

Accidents considered to fall within the "class 9" category, in particular, site-related events such as tornados, hurricanes, flood, or missile impacts, were not included in this assessment. Nor, for that matter, were accidental releases associated with radioactive waste management, including the storage of high level wastes at the reprocessing facility.

Comparisons with the risk from normal operations of the supporting fuel cycle and with occupational risks indicate that, on the basis of the annual operation of a 1000 MWe LWR, the risk from accidents in the supporting fuel cycle is orders of magnitude lower. On the same basis, the risk from accidents in the supporting fuel cycle is also slightly lower than that from reactor accidents, based upon the preliminary results of the Rasmussen study, and comparable to that from normal reactor operation.

Transportation accidents dominate the total accident risk, whereas the risk from accidents in mining, milling, and plutonium storage are relatively insignificant. The risks from the remaining components of the fuel cycle, with the exception of spent fuel reprocessing, are roughly of equal orders of magnitude. Moreover, the risks from accidents in uranium hexafluoride plants, enrichment facilities, uranium fuel fabrication, and mixed oxide fabrication plants, albeit small, are roughly comparable to those from normal operations.

A more comprehensive scoping analysis would include the risks associated with site-induced and other high consequence, low probability ("class 9") accidents. For example, although reprocessing plants and current designs for mixed oxide fabrication plants are hardened to withstand "design basis" natural disasters, the likelihood of exceeding the magnitude of these events and the attendant consequences should be assessed. Similarly, since the HEPA filters in these plants constitute an essential final barrier in the protection of the environment, a more detailed assessment of HEPA filter failure probabilities would appear warranted.

Accidents at the front end of the fuel cycle have the potential of releasing large quantities of uranium to the environment. The likelihood of a tornado dispersing nearly the entire inventory of uranium dioxide at a uranium fuel fabrication plant, for example, should be assessed.

Risks associated with proposed and postulated waste management alternatives have not been considered here, although this area is the subject of an extensive effort currently sponsored by the Energy Research and Development Administration. Nor have we assessed the likelihoods or consequences of accidents at the reprocessing facility involving the interim storage of high level wastes. In particular, the risk associated with a postulated loss-of-coolant to the liquid waste storage tanks should be assessed.

Accidental releases of chemicals at fuel cycle facilities also have the potential of producing environmental health effects. These include nitric acid and hydrogen fluoride, and in particular the HF chemically produced by the reaction of uranium hexafluoride with humid air. A more extensive scoping study should address the risks associated with these releases.

The accident probabilities adopted for this study are derived from two basic sources. For the front end of the fuel cycle, where considerable experience exists, incidents on record have been utilized, whenever possible, to derive accident likelihoods. Existing data compilations, however, are not comprehensive, and a more complete study would incorporate additional data extracted from ERDA and NRC compliance files, and from facility operating records, if made available.

For the back end of the fuel cycle, where little operating experience exists, we have relied heavily on past theoretical studies. Most of these, however, were also scoping investigations, and meagre resources have been devoted to investigating individual accident likelihoods on a realistic basis for actual design situations. In particular, more confidence could be placed in these results if accident scenarios associated with spent fuel reprocessing and transportation were examined in considerably more detail.

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APPENDIX A

50 YEAR DOSE COMMITMENT FACTORS FROM A ONE CURIE RELEASE ASSUMING A POPULATION DENSITY OF 160 PEOPLE/MILE^2

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** DOSE SUMMARY BY NUCLIDE-ALL PATHWAYS **


** POPULATION DOSE COMMITMENT (MANREM) **

NUCLIDE    T.BODY      GI-TRACT      BONE        LIVER       KIDNEY      THYROID      LUNG        SKIN
=========================================================================================================
H    3  !  9.04-04   ! 9.04-04    ! 9.04-04    !  9.04-04   !  9.04-04   ! 9.04-04   ! 9.04-04   !  9.04-04  !
        !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %   !  0.002%   !  0.002%   !
=========================================================================================================
C   14  ! 4.15E 01   ! 4.15E 01   ! 4.15E 01   ! 4.15E 01   ! 4.15E 01   ! 4.15E 01   ! 4.15E 01  ! 4.15E 01  !
        !  1.536%    !  2.212%    !  0.480%    !  1.894%    !  0.633%    !  0.296%    !  5.290%   !  5.931%   !
=========================================================================================================
NA  22  ! 1.05E 02   ! 8.21E 01   ! 6.70E 01   ! 6.70E 01   ! 6.70E 01   ! 6.70E 01   ! 6.85E 01  ! 6.70E 01  !
        !  3.873%    !  4.377%    !  0.776%    !  3.056%    !  1.022%    !  0.477%    !  8.735%   !  9.569%   !
=========================================================================================================
NA  24  ! 7.67E-02   ! 1.62E-01   ! 6.94E-02   ! 6.94E-02   ! 6.94E-02   ! 6.94E-02   ! 7.93E-02  ! 6.94E-02  !
        !  0.003%    !  0.009%    !  0.0  %    !  0.003%    !  0.001%    !  0.0  %    !  0.016%   !  0.010%   !
=========================================================================================================
AR  39  ! 6.93E-02   ! 6.93E-02   ! 6.93E-02   ! 6.93E-02   ! 6.93E-02   ! 6.93E-02   ! 3.99E-01  ! 2.52E 01  !
        !  0.003%    !  0.004%    !  0.0  %    !  0.003%    !  0.001%    !  0.0  %    !  0.051%   !  3.599%   !
=========================================================================================================
AR  41  ! 1.80E-04   ! 1.80E-04   ! 1.80E-04   ! 1.80E-04   ! 1.80E-04   ! 1.80E-04   ! 2.62E-04  ! 1.80E-04  !
        !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %   !  0.0  %   !
=========================================================================================================
CR  51  ! 4.35E-02   ! 1.42E-01   ! 3.98E-02   ! 3.98E-02   ! 4.09E-02   ! 4.42E-02   ! 4.94E-02  ! 3.98E-02  !
        !  0.002%    !  0.008%    !  0.0  %    !  0.002%    !  0.0  %    !  0.0  %    !  0.006%   !  0.006%   !
=========================================================================================================
MN  54  ! 7.94E 00   ! 1.07E 01   ! 7.81E 00   ! 8.58E 00   ! 7.81E 00   ! 7.81E 00   ! 8.26E 00  ! 7.81E 00  !
        !  0.294%    !  0.573%    !  0.090%    !  0.392%    !  0.119%    !  0.056%    !  1.053%   !  1.116%   !
=========================================================================================================
FE  55  ! 1.77E 00   ! 3.88E-01   ! 5.64E-01   ! 1.08E 00   ! 4.29E-02   ! 4.29E-02   ! 4.03E-01  ! 4.29E-02  !
        !  0.005%    !  0.021%    !  0.007%    !  0.049%    !  0.0  %    !  0.0  %    !  0.051%   !  0.006%   !
=========================================================================================================
FE  59  ! 2.46E 00   ! 9.65E 00   ! 2.02E 00   ! 4.13E 00   ! 1.62E 00   ! 1.62E 00   ! 2.51E 00  ! 1.62E 00  !
        !  0.091%    !  0.525%    !  0.030%    !  0.188%    !  0.025%    !  0.012%    !  0.320%   !  0.231%   !
=========================================================================================================
CO  58  ! 2.27E 00   ! 5.63E 00   ! 1.96E 00   ! 2.09E 00   ! 2.01E 00   ! 1.96E 00   ! 2.16E 00  ! 1.96E 00  !
        !  0.084%    !  0.306%    !  0.023%    !  0.095%    !  0.031%    !  0.014%    !  0.275%   !  0.280%   !
=========================================================================================================
CO  60  ! 1.46E 02   ! 1.56E 02   ! 1.44E 02   ! 1.44E 02   ! 1.44E 02   ! 1.44E 02   ! 1.45E 02  ! 1.44E 02  !
        !  5.393%    !  8.303%    !  1.661%    !  6.589%    !  2.196%    !  1.024%    ! 18.520%   ! 20.536%   !
=========================================================================================================
ZN  65  ! 1.76E 01   ! 2.42E 01   ! 1.23E 01   ! 4.42E 01   ! 2.43E 01   ! 4.26E 00   ! 4.49E 00  ! 4.26E 00  !
        !  0.651%    !  1.290%    !  0.142%    !  2.018%    !  0.370%    !  0.030%    !  0.572%   !  0.609%   !
=========================================================================================================
ZN  69M ! 7.48E-03   ! 1.07E-01   ! 8.93E-03   ! 1.01E-02   ! 9.28E-03   ! 7.59E-03   ! 1.13E-02  ! 7.59E-03  !
        !  0.0  %    !  0.006%    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.001%   !  0.001%   !
=========================================================================================================
ZN  69  ! 4.90E-06   ! 9.79E-06   ! 5.80E-05   ! 7.35E-05   ! 5.98E-05   ! 0.0        ! 1.47E-04  ! 0.0       !
        !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %    !  0.0  %   !  0.0  %   !
=========================================================================================================
BR 80   !  2.7-06    !  2.7-06    ! 2.7-06     !  2.7-06    !  2.7-06    ! 2.7-06     !  4.0-06   !  2.7-06   !
        !            !            !            !            !            !            !           !           !
---------------------------------------------------------------------------------------------------------
BR 80M  !  1.3-04    !  1.3-04    ! 1.3-04     !  1.3-04    !  1.3-04    ! 1.3-04     !  4.4-04      1.3-04   !
        !            !            !            !            !            !            !           !           !
---------------------------------------------------------------------------------------------------------
BR 82   !  8.5-02    !  8.5-02    ! 8.5-02     !  8.5-02    !  8.5-02    ! 8.5-02     !  9.3-02   !  8.5-02   !
```

BR 83	3.6-05	3.6-05	3.6-05	3.6-05	3.6-05	3.6-05	8.3-05	3.6-05
BR 84	1.1-04	1.1-04	1.1-04	1.1-04	1.1-04	1.1-04	1.2-04	1.1-04
KR 83M	1,42E-09 0,0 %	1,42E-09 0,0 %	1,42E-09 0,0 %	1,42E-09 0,0 %	1,42E-09 0,0 %	1,42E-09 0,0 %	5,61E-08 0,0 %	3,86E-07 0,0 %
KR 85M	5,25E-05 0,0 %	5,25E-05 0,0 %	5,25E-05 0,0 %	5,25E-05 0,0 %	5,25E-05 0,0 %	5,25E-05 0,0 %	5,33E-05 0,0 %	1,26E-04 0,0 %
KR 85	2,42E-04 0,0 %	2,42E-04 0,0 %	2,42E-04 0,0 %	2,42E-04 0,0 %	2,42E-04 0,0 %	2,42E-04 0,0 %	5,24E-04 0,0 %	2,05E-02 0,003%
KR 87	7,64E-05 0,0 %	7,64E-05 0,0 %	7,64E-05 0,0 %	7,64E-05 0,0 %	7,64E-05 0,0 %	7,64E-05 0,0 %	7,77E-05 0,0 %	1,01E-03 0,0 %
KR 88	4,24E-04 0,0 %	4,24E-04 0,0 %	4,24E-04 0,0 %	4,26E-04 0,0 %	4,24E-04 0,0 %	4,24E-04 0,0 %	4,24E-04 0,0 %	5,89E-04 0,0 %
KR 89	9,04E-06 0,0 %	9,04E-06 0,0 %	9,04E-06 0,0 %	9,04E-06 0,0 %	9,04E-06 0,0 %	9,04E-06 0,0 %	9,04E-06 0,0 %	1,57E-05 0,0 %
KR 90	1,47E-06 0,0 %	1,47E-06 0,0 %	1,47E-06 0,0 %	1,47E-06 0,0 %	1,47E-06 0,0 %	1,47E-06 0,0 %	1,47E-06 0,0 %	2,37E-06 0,0 %
RB 86	4,31E 00 0,159%	1,92E 00 0,102%	4,97E-02 0,0 %	9,00E 00 0,411%	4,97E-02 0,0 %	4,97E-02 0,0 %	2,21E-01 0,028%	4,97E-02 0,007%
RB 87	1,18E 01 0,437%	2,38E 00 0,127%	0,0 0,0 %	3,54E 01 1,617%	0,0 0,0 %	0,0 0,0 %	2,40E-01 0,031%	0,0 0,0 %
RB 89	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05
SR 89	2,66E-01 0,010%	7,10E 00 0,378%	1,27E 01 0,146%	0,0 0,0 %	0,0 0,0 %	0,0 0,0 %	3,87E-01 0,049%	0,0 0,0 %
SR 90	5,21E 01 1,927%	8,41E 00 0,448%	2,56E 03 29,573%	0,0 0,0 %	0,0 0,0 %	0,0 0,0 %	2,39E 00 0,305%	0,0 0,0 %
SR 91	2,38E-03 0,0 %	3,50E-02 0,002%	1,38E-02 0,0 %	1,88E-03 0,0 %	1,88E-03 0,0 %	1,88E-03 0,0 %	9,26E-03 0,001%	1,88E-03 0,0 %
Y 90	1,01E-03 0,0 %	3,92E-01 0,021%	3,03E-02 0,0 %	0,0 0,0 %	0,0 0,0 %	0,0 0,0 %	3,03E-02 0,004%	0,0 0,0 %
Y 91M	5,91E-04 0,0 %	1,07E-03 0,0 %	8,73E-04 0,0 %	5,79E-04 0,0 %	5,79E-04 0,0 %	5,79E-04 0,0 %	8,73E-04 0,0 %	5,79E-04 0,0 %
Y 91	2,14E-02 0,0 %	7,29E 00 0,384%	8,05E-01 0,009%	4,64E-03 0,0 %	4,64E-03 0,0 %	4,64E-03 0,0 %	3,98E-01 0,051%	4,64E-03 0,0 %
Y 92	6,02E-04 0,0 %	2,99E-02 0,002%	3,45E-03 0,0 %	5,04E-04 0,0 %	5,04E-04 0,0 %	5,04E-04 0,0 %	3,45E-03 0,0 %	5,04E-04 0,0 %
ZR 95	2,95E 00 0,109%	1,42E 01 0,756%	3,06E 00 0,035%	2,97E 00 0,135%	2,99E 00 0,046%	2,92E 00 0,021%	3,34E 00 0,425%	2,92E 00 0,417%
ZR 97	1,92E-02 0,0 %	1,19E-01 0,006%	2,46E-02 0,0 %	1,97E-02 0,0 %	2,02E-02 0,0 %	1,87E-02 0,0 %	3,35E-02 0,004%	1,87E-02 0,003%
NB 95	7,94E-01 0,029%	1,96E 01 1,046%	8,16E-01 0,009%	8,01E-01 0,037%	8,01E-01 0,012%	7,88E-01 0,006%	9,09E-01 0,116%	7,88E-01 0,112%

```
NB  97  | 1,06E-03 | 2,51E-03 | 1,24E-03 | 1,09E-03 | 1,09E-03 | 1,04E-03 | 1,86E-03 | 1,04E-03 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
TC  97M | 1,97E-01 | 1,48E 01 | 2,36E-01 | 5,91E-01 | 5,91E 00 | 0,0      | 5,91E-02 | 3,94E-01 |
        |  0,007%  |  0,787%  |  0,003%  |  0,027%  |  0,090%  |  0,0   % |  0,008%  |  0,056%  |
------------------------------------------------------------------------------------------------
TC  97  | 1,44E-01 | 1,29E 01 | 1,29E-01 | 5,17E-01 | 6,47E 00 | 0,0      | 9,12E-02 | 2,59E-02 |
        |  0,005%  |  0,689%  |  0,001%  |  0,028%  |  0,099%  |  0,0   % |  0,012%  |  0,004%  |
------------------------------------------------------------------------------------------------
TC  99M | 1,01E-03 | 1,59E-03 | 9,96E-04 | 9,96E-04 | 1,01E-03 | 9,96E-04 | 1,09E-03 | 9,96E-04 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
TC  99  | 8,62E-01 | 8,62E 01 | 1,72E 00 | 2,59E 00 | 3,23E 01 | 0,0      | 4,99E-01 | 2,59E 00 |
        |  0,032%  |  4,595%  |  0,020%  |  0,118%  |  0,493%  |  0,0   % |  0,066%  |  0,370%  |
------------------------------------------------------------------------------------------------
MO  99  | 3,07E-02 | 2,33E-01 | 1,52E-02 | 9,01E-02 | 2,00E-01 | 1,52E-02 | 3,03E-02 | 1,52E-02 |
        |  0,001%  |  0,012%  |  0,0   % |  0,004%  |  0,003%  |  0,0   % |  0,004%  |  0,002%  |
------------------------------------------------------------------------------------------------
RU 103  | 7,34E-01 | 3,48E 01 | 8,96E-01 | 6,19E-01 | 1,54E 00 | 6,19E-01 | 7,44E-01 | 6,19E-01 |
        |  0,027%  |  1,854%  |  0,010%  |  0,028%  |  0,024%  |  0,004%  |  0,095%  |  0,088%  |
------------------------------------------------------------------------------------------------
RH 103M | 4,58E-06 | 1,51E-04 | 5,56E-06 | 5,07E-06 | 1,09E-05 | 3,60E-06 | 3,30E-05 | 3,60E-06 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
RU 106  | 1,05E 00 | 6,29E 02 | 1,26E 01 | 0,0      | 1,50E 01 | 0,0      | 2,28E 00 | 0,0      |
        |  0,039%  | 33,536%  |  0,146%  |  0,0   % |  0,281%  |  0,0   % |  0,290%  |  0,0   % |
------------------------------------------------------------------------------------------------
AG 110M | 2,13E 01 | 1,73E 02 | 2,56E 01 | 2,15E 01 | 2,18E 01 | 2,10E 01 | 2,25E 01 | 2,10E 01 |
        |  0,786%  |  9,198%  |  0,296%  |  0,981%  |  0,333%  |  0,150%  |  2,870%  |  3,004%  |
------------------------------------------------------------------------------------------------
SB 125  | 1,56E 01 | 3,97E 01 | 1,72E 01 | 1,54E 01 | 1,55E 01 | 1,56E 01 | 1,57E 01 | 1,55E 01 |
        |  0,577%  |  2,117%  |  0,199%  |  0,710%  |  0,237%  |  0,111%  |  1,997%  |  2,217%  |
------------------------------------------------------------------------------------------------
TE 127  | 1,00E-04 | 9,98E-03 | 3,48E-04 | 1,50E-04 | 7,98E-04 | 1,99E-04 | 1,03E-03 | 5,07E-05 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
TE 127M | 7,97E-01 | 1,61E 01 | 6,49E 00 | 2,65E 00 | 2,14E 01 | 1,37E 00 | 5,04E-01 | 8,44E-02 |
        |  0,029%  |  0,857%  |  0,075%  |  0,121%  |  0,327%  |  0,010%  |  0,064%  |  0,012%  |
------------------------------------------------------------------------------------------------
TE 129  | 1,45E-06 | 3,06E-03 | 1,94E-06 | 1,63E-04 | 4,15E-04 | 1,00E-04 | 5,41E-04 | 1,21E-04 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
TE 129M | 1,14E 00 | 2,17E 01 | 4,40E 00 | 2,23E 00 | 1,63E 01 | 1,69E 00 | 4,28E-01 | 6,04E-02 |
        |  0,042%  |  1,156%  |  0,051%  |  0,102%  |  0,249%  |  0,012%  |  0,055%  |  0,009%  |
------------------------------------------------------------------------------------------------
TE 131M | 7,52E-02 | 1,57E-01 | 6,92E-02 | 6,91E-02 | 8,11E-02 | 3,32E 00 | 8,21E-02 | 6,69E-02 |
        |  0,003%  |  0,008%  |  0,0   % |  0,003%  |  0,001%  |  0,024%  |  0,010%  |  0,010%  |
------------------------------------------------------------------------------------------------
TE 132  | 2,73E-01 | 1,48E 00 | 2,88E-01 | 2,80E-01 | 4,51E-01 | 4,88E 00 | 2,91E-01 | 2,47E-01 |
        |  0,010%  |  0,079%  |  0,003%  |  0,013%  |  0,007%  |  0,035%  |  0,037%  |  0,035%  |
------------------------------------------------------------------------------------------------
XE 131M | 2,63E-04 | 2,63E-04 | 2,63E-04 | 2,63E-04 | 2,63E-04 | 2,63E-04 | 2,95E-04 | 1,86E-03 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
XE 133M | 1,39E-04 | 1,39E-04 | 1,39E-04 | 1,39E-04 | 1,39E-04 | 1,39E-04 | 1,47E-04 | 7,45E-04 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
XE 133  | 3,80E-04 | 3,80E-04 | 3,80E-04 | 3,80E-04 | 3,80E-04 | 3,80E-04 | 3,92E-04 | 8,95E-04 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
XE 135M | 8,27E-06 | 8,27E-06 | 8,27E-06 | 8,27E-06 | 8,27E-06 | 8,27E-06 | 8,30E-06 | 1,17E-05 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
------------------------------------------------------------------------------------------------
XE 135  | 1,31E-04 | 1,31E-04 | 1,31E-04 | 1,31E-04 | 1,31E-04 | 1,31E-04 | 1,72E-04 | 3,73E-04 |
        |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |  0,0   % |
```

XE 137	9,45E-07 0,0 %	9,45E-07 0,0 %	9,45E-07 0,0 %	9,45E-07 0,0 %	9,45E-07 0,0 %	9,45E-07 0,0 %	1,02E-06 0,0 %	9,20E-06 0,0 %
XE 138	2,52E-05 0,0 %	2,52E-05 0,0 %	2,52E-05 0,0 %	2,52E-05 0,0 %	2,52E-05 0,0 %	2,52E-05 0,0 %	2,52E-05 0,0 %	4,04E-05 0,0 %
I 128	9.2-05	9.2-05	9.2-05	9.2-05	9.2-05	3.8-04	9.6-05	9.2-05
I 129	3,10E 01 1,250%	1,51E 01 0,803%	1,02E 01 0,164%	1,02E 01 0,440%	1,02E 01 0,217%	1,31E 04 93,315%	1,05E 01 1,042%	1,02E 01 2,030%
I 130	1.6-02	1.6-02	1.6-02	1.6-02	1.6-02	1.3-01	1.8-02	1.6-02
I 131	6,43E-01 0,024%	4,21E-01 0,022%	9,62E-02 0,001%	9,62E-02 0,004%	9,62E-02 0,001%	2,69E 02 1,914%	1,20E-01 0,016%	9,62E-02 0,014%
I 132	6,86E-03 0,0 %	1,05E-02 0,0 %	6,70E-03 0,0 %	6,70E-03 0,0 %	6,70E-03 0,0 %	8,02E-02 0,0 %	8,17E-03 0,001%	6,70E-03 0,0 %
I 133	2,10E-02 0,0 %	2,59E-02 0,001%	1,83E-02 0,0 %	1,83E-02 0,0 %	1,83E-02 0,0 %	1,57E 00 0,011%	2,57E-02 0,003%	1,83E-02 0,003%
I 134	2.6-04	2.6-04	2.6-04	2.6-04	2.6-04	1.8-03	2.8-04	2.6-04
I 135	1,50E-02 0,0 %	4,43E-02 0,002%	1,47E-02 0,0 %	1,47E-02 0,0 %	1,47E-02 0,0 %	1,47E-01 0,001%	1,70E-02 0,002%	1,47E-02 0,002%
CS 134M	8,87E-03 0,0 %	7,34E-03 0,0 %	8,15E-03 0,0 %	1,42E-02 0,0 %	7,86E-03 0,0 %	5,70E-03 0,0 %	6,91E-03 0,0 %	5,70E-03 0,0 %
CS 134	8,70E 01 3,249%	3,83E 01 2,030%	7,58E 01 0,876%	1,77E 02 8,064%	7,07E 01 1,079%	3,54E 01 0,252%	5,00E 01 6,055%	3,54E 01 5,053%
CS 136	1,07E 02 3,954%	1,20E 00 0,004%	1,39E 00 0,016%	2,20E 00 0,103%	1,78E 00 0,027%	8,13E-01 0,006%	1,01E 00 0,129%	8,13E-01 0,116%
CS 137	1,44E 02 5,119%	1,20E 02 4,360%	1,80E 02 2,084%	1,94E 02 8,951%	1,49E 02 2,272%	1,17E 02 0,837%	1,20E 02 16,096%	1,17E 02 16,786%
CS 138	1.3-04	1.3-04	1.3-04	1.3-04	1.3-04	1.3-04	1.4-04	1.3-04
BA 140	1,05E-02 0,0 %	2,80E 00 0,149%	9,81E-01 0,011%	1,08E-03 0,0 %	3,45E-04 0,0 %	7,09E-06 0,0 %	3,31E-01 0,042%	7,09E-06 0,0 %
LA 140	1,19E-01 0,004%	2,50E-01 0,013%	1,25E-01 0,001%	1,22E-01 0,006%	1,18E-01 0,002%	1,18E-01 0,0 %	1,48E-01 0,019%	1,18E-01 0,017%
CE 141	8,58E-02 0,003%	1,92E 00 0,102%	1,20E-01 0,001%	1,01E-01 0,005%	9,45E-02 0,001%	8,23E-02 0,0 %	1,58E-01 0,020%	8,23E-02 0,012%
CE 143	1,46E-02 0,0 %	1,08E-01 0,006%	2,04E-02 0,0 %	1,85E-02 0,0 %	1,57E-02 0,0 %	1,41E-02 0,0 %	2,96E-02 0,004%	1,41E-02 0,002%
CE 144	5,92E-01 0,022%	3,25E 01 1,732%	3,52E 00 0,041%	1,54E 00 0,073%	1,10E 00 0,017%	4,39E-01 0,003%	2,69E 00 0,342%	4,39E-01 0,063%
PR 143	1,88E-03 0,0 %	1,70E 00 0,091%	3,37E-02 0,0 %	1,80E-02 0,0 %	8,41E-03 0,0 %	0,0 0,0 %	5,55E-02 0,007%	0,0 0,0 %
PM 147	1,62E-02 0,0 %	1,78E 00 0,095%	4,82E-01 0,006%	4,80E-02 0,002%	4,07E-02 0,001%	0,0 0,0 %	1,58E-01 0,020%	0,0 0,0 %

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Pm 147	1,62E-02 0,0 %	1,70E 00 0,095%	4,62E-01 0,006%	9,80E-02 0,002%	6,87E-02 0,001%	0,0 0,0 %	1,56E-01 0,020%	0,0 0,0 %
Pm 149	5,04E-04 0,0 %	1,22E-01 0,007%	6,03E-03 0,0 %	1,01E-03 0,0 %	1,51E-03 0,0 %	1,39E-06 0,0 %	1,00E-02 0,001%	1,35E-06 0,0 %
Nd 147	5,51E-02 0,002%	1,10E 00 0,063%	7,70E-02 0,0 %	6,63E-02 0,006%	6,99E-02 0,001%	9,32E-02 0,0 %	9,66E-02 0,012%	9,32E-02 0,006%
Sm 151	2,90E-02 0,001%	9,75E-01 0,052%	5,08E-01 0,006%	[illegible] 0,006%	[illegible] 0,001%	[illegible] 0,0 %	1,00E-01 0,014%	[illegible] 0,002%
Sm 153	2,13E-04 0,0 %	5,95E-02 0,005%	3,02E-03 0,0 %	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Eu 154	1,49E 02 7,002%	2,00E 02 11,058%	1,99E 02 2,296%	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Eu 155	1,48E 00 0,055%	3,20E 00 0,171%	1,77E 00 0,020%	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Pb 210	1,70E 03 62,002%	2,38E 01 1,270%	5,25E 03 60,677%	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

```
** DOSE SUMMARY BY NUCLIDE-ALL PATHWAYS **

** POPULATION DOSE COMMITMENT (MANREM) **

NUCLIDE   T.BODY      GI-TRACT     BONE        LIVER       KIDNEY      THYROID     LUNG        SKIN
-----------------------------------------------------------------------------------------------------
BI 210  | 2.05E-01  | 9.86E-01  | 9.53E-01  | 1.70E 00  | 7.92E 00  | 0.0       | 2.01 01   | 0.0       |
        |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |  0.015%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
BI 212  | 1.38E-03  | 3.83E-03  | 2.11E-03  | 7.50E-03  | 9.81E-02  | 1.50E-04  | 1.35 00   | 1.50E-04  |
        |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |
-----------------------------------------------------------------------------------------------------
PB 212  | 2.83E-02  | 5.45E-02  | 2.01E-01  | 5.32E-02  | 4.90E-01  | 3.80E-03  | 1.36 01   | 3.80E-03  |
        |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |  0.003%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
PB 210  | 1.74E 01  | 1.38E 01  | 8.45E 01  | 1.41E 02  | 5.24E 02  | 0.0       | 1.76 01   | 0.0       |
        |  0.001%   |  0.003%   |   0.0  %  |  0.010%   |  0.007%   |   0.0  %  |  0.013%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
RN 222  | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
        | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
-----------------------------------------------------------------------------------------------------
[illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
        | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
-----------------------------------------------------------------------------------------------------
RA 223  | 1.78E 03  | 9.30E 01  | 7.93E 03  | 0.0       | 0.0       | 0.0       | 1.31 02   | 0.0       |
        | [illegible] | [illegible] | [illegible] |   0.0  %  |   0.0  %  |   0.0  %  | [illegible] |   0.0  %  |
-----------------------------------------------------------------------------------------------------
RA 224  | 7.52E 00  | 9.57E 00  | 6.75E 01  | 3.50E-02  | 9.79E-02  | 3.20E-02  | 1.97 02   | 3.20E-02  |
        |   0.0  %  |  0.002%   |   0.0  %  |   0.0  %  |   0.0  %  |  0.002%   |  0.708%   |  0.002%   |
-----------------------------------------------------------------------------------------------------
RA 226  | 2.79E 04  | 5.03E 01  | 2.06E 04  | 3.33E 02  | 1.53E 03  | 4.73E 00  | 2.96 01   | 4.73E 00  |
        |  2.279%   |  0.009%   |  0.084%   |  0.023%   |  0.022%   |  0.280%   |  0.039%   |  0.280%   |
-----------------------------------------------------------------------------------------------------
RA 228  | 1.35E 04  | 1.22E 02  | 1.07E 05  | 8.20E 01  | 8.73E 01  | 8.06E 01  | 1.72 04   | 8.06E 01  |
        |  1.100%   |  0.023%   |  0.339%   |  0.006%   |  0.001%   |  4.836%   |  4.376%   |  4.836%   |
-----------------------------------------------------------------------------------------------------
AC 227  | 6.64E 02  | 2.13E 01  | 1.36E 04  | 1.76E 03  | 6.00E 02  | 0.0       | 1.82 04   | 0.0       |
        |  0.054%   |  0.004%   |  0.043%   |  0.122%   |  0.008%   |   0.0  %  |  4.664%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
AC 228  | 3.98E-02  | 2.84E-02  | 6.44E-01  | 1.13E-01  | 5.50E-02  | 8.03E-03  | 2.38 01   | 8.03E-03  |
        |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |  0.006%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
TH 227  | 2.85E 01  | 9.94E 00  | 1.59E 02  | 6.68E-01  | 5.80E 00  | 0.0       | 1.26 02   | 0.0       |
        |  0.002%   |  0.002%   |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |  0.038%   |   0.0  %  |
-----------------------------------------------------------------------------------------------------
TH 228  | 9.03E 01  | 3.94E 01  | 3.22E 03  | 5.84E 01  | 2.40E 02  | 6.52E 00  | 6.68 03   | 6.52E 00  |
        |  0.007%   |  0.007%   |  0.010%   |  0.004%   |  0.003%   |  0.392%   | 16.507%   |  0.392%   |
-----------------------------------------------------------------------------------------------------
TH 230  | 3.38E 02  | 1.19E 01  | 1.23E 04  | 7.06E 02  | 2.52E 03  | 1.55E 00  | 3.20 03   | 1.55E 00  |
        |  0.048%   |  0.002%   |  0.039%   |  0.049%   |  0.036%   |  0.093%   | 11.318%   |  0.093%   |
-----------------------------------------------------------------------------------------------------
TH 231  | 7.88E-02  | 2.80E-02  | 1.19E 00  | 5.28E-02  | 2.31E-01  | 3.35E-03  | 6.33 -03  | 3.35E-03  |
        |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |   0.0  %  |
-----------------------------------------------------------------------------------------------------
TH 232  | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
        | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
```

TH 234	2.20E 00 0.0 %	7.97E 00 0.001%	2.55E 00 0.0 %	2.21E 00 0.0 %	2.31E 00 0.0 %	2.19E 00 0.131%	2.90 00 0.010%	2.19E 00 0.131%
PA 230	1.06E 02 0.012%	1.31E 04 2.009%	9.26E 03 0.017%	0.0 0.0 %	6.90E 02 0.009%	0.0 0.0 %	3.03 02 0.000%	0.0 0.0 %
PA 231	1.17E 06 95.792%	5.09E 05 87.359%	3.13E 07 98.060%	1.41E 06 97.004%	7.00E 06 99.359%	0.0 0.0 %	3.97 03 0.075%	0.0 0.0 %
PA 233	2.08E 00 0.0 %	3.00E 04 7.169%	1.14E 01 0.0 %	2.14E 00 0.0 %	7.90E 00 0.0 %	2.18E-01 0.013%	2.11 00 0.002%	2.18E-01 0.013%
PA 234M	5.15E-04 0.0 %	6.70E-04 0.0 %	2.35E-04 0.0 %	5.67E-04 0.0 %	6.70E-04 0.0 %	6.70E-04 0.0 %	2.94-01 0.0 %	6.70E-04 0.0 %
PA 234	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
U [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
U [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
U [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
U 234	1.50E 01 0.001%	1.30E 01 0.002%	7.23E 02 0.0 %	1.19E 00 0.0 %	9.03E 01 0.0 %	1.19E 00 0.071%	3.86 03 0.956%	1.19E 00 0.071%
U 235	1.30E 02 0.011%	1.40E 02 0.022%	6.43E 02 0.002%	1.19E 02 0.008%	2.94E 02 0.004%	1.03E 02 6.164%	3.94 03 1.572%	1.03E 02 6.164%
U 236	1.52E 01 0.001%	1.26E 01 0.002%	1.09E 02 0.0 %	7.96E-01 0.0 %	6.79E 01 0.0 %	7.96E-01 0.048%	3.84 03 0.853%	7.96E-01 0.048%
U 237	6.18E-02 0.0 %	6.44E-02 0.0 %	6.25E-02 0.0 %	6.08E-02 0.0 %	6.91E-02 0.0 %	6.08E-02 0.004%	1.47-01 0.0 %	6.08E-02 0.004%
U 238	1.52 01 0.476%	1.26 01 0.179%	1.09 02 0.000%	7.96-01 0.066%	7.51 01 0.024%	7.96-01 93.446%	3.08 03 7.172%	7.96-01 93.446%
U 239	6.9-06	6.9-06	7.5-06	6.9-06	6.9-06	6.9-06	1.2-05	6.9-06
NP 237	3.91E 02 0.012%	1.44E 04 2.000%	9.43E 03 0.031%	7.30E 02 0.051%	2.37E 04 0.030%	1.13E 02 0.791%	3.95 03 1.000%	1.13E 02 0.791%
NP 239	1.08E-02 0.0 %	8.50E-02 0.0 %	1.09E-02 0.0 %	1.09E-02 0.0 %	1.00E-02 0.0 %	1.03E-02 0.0 %	1.50-01 0.0 %	1.03E-02 0.0 %
PU 238	1.23E 02 [illegible]	1.09E 01 [illegible]	1.29E 02 [illegible]	1.21E 03 [illegible]	1.21E 03 [illegible]	6.92E-01 0.052%	3.86 03 [illegible]	6.92E-01 0.052%
PU 239	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
PU 240	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
PU 241	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

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PU 242	3,20E 02 0,026%	1,04E 01 0,002%	1,01E 00 0,051%	1,02E 03 0,118%	1,01E 03 0,023%	6,75E-01 0,040%	3.0E 03 3.39%	6,75E-01 0,040%
PU 243	3,93E-05 0,0 %	3,09E-03 0,0 %	1,32E-03 0,0 %	1,09E-00 0,0 %	2,21E-00 0,0 %	5,59E-06 0,0 %	5,60E-06 0,0 %	5,59E-06 0,0 %
PU 244	4,00E 02 0,033%	2,91E 01 0,005%	1,02E 00 0,051%	0,0 0,0 %	1,02E 03 0,023%	0,0 0,0 %	3.0E 03 3.30%	0,0 0,0 %
AM 241	3,40E 02 0,026%	2,79E 01 0,005%	4,94E 03 0,016%	1,44E 03 0,115%	2,49E 03 0,035%	1,00E 01 1,003%	3.06 03 0,975%	1,00E 01 1,003%
AM 242M	3,30E 02 0,027%	3,27E 00 0,0 %	4,75E 03 0,016%	1,44E 03 0,114%	2,20E 03 0,035%	0,0 0,0 %	3.71 03 0,276%	0,0 0,0 %
AM 242	1,65E-02 0,0 %	1,49E-02 0,0 %	1,77E-01 0,0 %	[illegible]	[illegible]	[illegible]	3.76 03 0,061%	[illegible]
AM 243	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
CM 242	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	0,0 0,0 %	[illegible]	0,0 0,0 %
CM 243	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
CM 244	2,27E 02 0,017%	1,80E 01 0,002%	4,71E 03 0,016%	1,40E 03 0,113%	1,02E 03 0,017%	3,71E 02 0,850%	3.72 03 1.72%	3,71E 02 0,850%
CM 245	1,80E 02 0,015%	1,21E 01 0,002%	3,27E 03 0,010%	1,23E 03 0,095%	9,27E 02 0,012%	2,20E 00 0,135%	2.09 03 1.72%	2,20E 00 0,135%
CM 246	3,31E 02 0,027%	1,01E 01 0,002%	4,96E 03 0,016%	1,05E 03 0,110%	1,05E 03 0,024%	0,0 0,0 %	3.84 03 0,840%	0,0 0,0 %
CM 247	3,31E 02 0,027%	1,53E 01 0,003%	4,96E 03 0,016%	1,05E 03 0,110%	1,05E 03 0,023%	1,37E-01 0,008%	3.86 03 0,849%	1,37E-01 0,008%
CM 248	2,77E 04 0,226%	3,03E 02 0,056%	4,96E 04 0,157%	1,24E 04 0,856%	1,24E 04 0,170%	0,0 0,0 %	3.86 03 8,478%	0,0 0,0 %
CM 249	9,79E-05 0,0 %	7,35E-06 0,0 %	1,07E-03 0,0 %	0,0 0,0 %	0,0 0,0 %	0,0 0,0 %	4.70-03 0,0 %	0,0 0,0 %

APPENDIX B

SOURCE TERMS FROM NORMAL OPERATIONS

B .1 Normal Operations Source Term from Mining

To Atmosphere

Rn 222	1.3×10^{4} Ci/yr

B .2 Normal Operations Source Term from Milling

To Atmosphere

Nuclide	Activity Released (Ci/yr)
U 238	6.2×10^{-2}
U 234	6.2×10^{-2}
U 235	2.8×10^{-3}
Th 234	3.3×10^{-3}
Th 230	8.8×10^{-3}
Ra 226	5.9×10^{-3}
Rn 222	2.2×10^{3}

B.3 Normal Operations Source Term from UF_6 Conversion

To Atmosphere

Nuclide	Activity Released (Ci/yr)
U 234	1.6×10^{-2}
U 235	6.6×10^{-4}
U 238	1.6×10^{-2}
Th 230	4.0×10^{-4}
Th 234	1.0×10^{-2}
Ra 226	4.6×10^{-5}
Rn 222	1.6×10^{-4}

To Watercourse

U 234	5.5×10^{-1}
U 235	2.5×10^{-2}
U 238	5.5×10^{-1}
Th 230	1.8×10^{-1}
Th 234	5.1×10^{-1}
Ra 226	5.3×10^{-3}

B.4 Normal Operations Source Term from Enrichment

To Atmosphere

Nuclide	Activity Released (Ci/yr)
U 232	3.4×10^{-3}
U 233	7.6×10^{-6}
U 234	1.2×10^{-1}
U 235	5.8×10^{-3}
U 236	9.4×10^{-3}
U 238	1.7×10^{-2}

To Watercourse

U 232	.19
U 233	.00042
U 234	6.36
U 235	.19
U 236	.52
U 238	.95

B.5 Normal Operations Source Term from Uranium Fuel Fabrication

To Atmosphere

Nuclide	Activity Released (Ci/yr)
U 234	4.7×10^{-3}
U 235	1.3×10^{-4}
U 236	2.0×10^{-4}
U 238	5.8×10^{-4}
Th 231	1.3×10^{-4}
Th 234	5.8×10^{-4}

To Watercourse

U 234	1.1
U 235	3.2×10^{-2}
U 236	4.7×10^{-2}
U 238	1.4×10^{-1}
Th 231	3.2×10^{-2}
Th 234	1.4×10^{-1}

B.6 Normal Operations Source Term from Reprocessing

To Atmosphere

Nuclide	Activity Released (Ci/yr)
H 3	1.10×10^{6}
C 14	7.00×10^{2}
Kr 85	1.50×10^{7}
I 129	2.90
I 131	3.50×10^{1}
Ru 103	1.10
Ru 106	6.60
Sr 89	2.50×10^{-1}
Sr 90	2.20×10^{-1}
Y 90	2.20×10^{-1}
Y 91	4.20×10^{-1}
Zr 95	7.50×10^{-1}
Nb 95	1.40
Ag 110m	8.40×10^{-3}
Sb 125	2.60×10^{-2}
Te 127m	1.80×10^{-2}
Te 129m	6.60×10^{-3}
Cs 134	6.30×10^{-1}
Cs 137	3.20×10^{-1}
Ce 141	1.40×10^{-1}
Ce 144	2.20

To Atmosphere

(continued)

Nuclide	Activity Released (Ci/yr)
Pm 147	2.90×10^{-1}
Eu 154	2.10×10^{-2}
Eu 155	2.00×10^{-2}
U 232	3.20×10^{-4}
U 234	2.60×10^{-2}
U 235	4.80×10^{-4}
U 236	1.10×10^{-2}
U 237	1.30×10^{-1}
U 238	9.20×10^{-3}
Pu 238	5.20×10^{-2}
Pu 239	2.80×10^{-3}
Pu 240	5.10×10^{-3}
Pu 241	1.30
Pu 242	2.80×10^{-5}
Am 241	1.10×10^{-3}
Am 243	2.60×10^{-4}
Cm 242	1.40×10^{-1}
Cm 244	5.70×10^{-2}

B .7 Normal Operations Source Term from Mixed Oxide Fuel Fabrication

To Atmosphere

Nuclide	Activity Released (Ci/yr)
Pu 238	4.2×10^{-3}
Pu 239	2.4×10^{-4}
Pu 240	4.2×10^{-4}
Pu 241	8.8×10^{-2}
Pu 242	2.2×10^{-6}

To Watercourse

Pu 238	2.7×10^{-2}
Pu 239	1.5×10^{-3}
Pu 240	2.7×10^{-3}
Pu 241	5.6×10^{-1}
Pu 242	1.4×10^{-4}

B.8 Normal Operations Source Term from Transportation

No radiological effluents to the enviornment. Direct shine doses to the general public scaled from results given in Reference 16.

B.9 Normal Operations Source Term from Reactor*

To Atmosphere

Nuclide	Activity Released (Ci/yr)
Kr 83m	3.3
Kr 85m	35
Kr 85	590
Kr 87	13
Kr 88	22
Kr 89	40
Xe 131m	57
Xe 133m	15
Xe 133	1800
Xe 135m	17
Xe 135	44
Xe 137	84
Xe 138	73
I 131	.13
I 133	.52

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* Assumed to be comprised of 2/3 PWR's plus 1/3 BWR's.

To Watercourse

Nuclide	Activity Released (Ci/yr)
Corrosion and Activation Products	
^{24}Na	.0001
^{32}P	.00002
^{33}P	.0001
^{51}Cr	.0003
^{54}Mn	.0001
^{56}Mn	.00004
^{55}Fe	.001
^{59}Fe	.0004
^{58}Co	.005
^{60}Co	.0004
^{63}Ni	.00002
^{92}Nb	.00008
^{117m}Sn	.00002
^{121}Sn	--
^{185}W	.00001
^{187}W	.0008
^{237}U	--
^{239}Np	.0003
^{241}Pu	.00001
Fission Products	
^{82}Br	.00002
^{86}Rb	--

To Watercourse

(continued)

Nuclide	Activity Released (Ci/yr)
Fission Products (continued)	
^{89}Sr	.0025
^{90}Sr	.00014
^{90}Y	.004
^{91}Sr	.0009
^{91m}Y	.0006
^{91}Y	.11
^{92}Sr	--
^{92}Y	.005
^{93}Y	.066
^{95}Zr	.00004
^{95}Nb	.00003
^{97}Zr	.00001
^{97m}Nb	.00001
^{97}Nb	.00001
^{99}Mo	.04
^{99m}Tc	.04
^{103}Ru	.00003
^{103m}Rh	.00003
^{105}Rh	.00002
^{106}Ru	.00001
^{106}Rh	.00001
^{127m}Te	.00001

To Watercourse
(continued)

Nuclide	Activity Released (Ci/yr)
Fission Products (continued)	
$^{127}\mathrm{Te}$	.00002
$^{129m}\mathrm{Te}$	.00006
$^{129}\mathrm{Te}$	.00003
$^{130}\mathrm{I}$	.0001
$^{131m}\mathrm{Te}$	.00006
$^{131}\mathrm{Te}$	.00002
$^{131}\mathrm{I}$	.08
$^{132}\mathrm{Te}$	.0007
$^{132}\mathrm{I}$	.0014
$^{133}\mathrm{I}$	.026
$^{134}\mathrm{Cs}$	.0026
$^{135}\mathrm{I}$	.0041
$^{136}\mathrm{Cs}$	.0012
$^{137}\mathrm{Cs}$	.0018
$^{137m}\mathrm{Ba}$	.0017
$^{140}\mathrm{Ba}$	.004
$^{140}\mathrm{La}$	.0025
$^{141}\mathrm{La}$	.00003
$^{141}\mathrm{Ce}$	.00009
$^{143}\mathrm{Ce}$	.00005
$^{143}\mathrm{Pr}$	.00003

To Watercourse

(continued)

Nuclide	Activity Released (Ci/yr)
Fission Products (continued)	
^{144}Ce	.00002
^{144}Pr	.00002
^{147}Nd	.00001
^{149}Pm	.00001
^{153}Sm	--
Tritium	240
All Others	.00005